Access DB# 70528

SEARCH REQUEST FORM

Scientific and Technical Information Center

Requester's Full Name: Harry Wilkins Examiner # : 78403 Date: 7/9/02
Art Unit: 1742 Phone Number 305-9927 Serial Number: 09/773,782
Mail Box and Bldg/Room Location: CP3-7326 Results Format Preferred (circle): PAPER DISK E-MAIL

If more than one search is submitted, please prioritize searches in order of need.

****

Please provide a detailed statement of the search topic, and describe as specifically as possible the subject matter to be searched. Include the elected species or structures, keywords, synonyms, acronyms, and registry numbers, and combine with the concept or utility of the invention. Define any terms that may have a special meaning. Give examples or relevant citations, authors, etc, if known. Please attach a copy of the cover sheet, pertinent claims, and abstract.

Title of Invention: Creep Resistant Zirconium Alloy and Nuclear Fuel Cladding Incorporating Said Alloy

Inventors (please provide full names): Raymond Grant Rowe, Ronald Bert Adamson, Sheikh Tahir Mahmood

Earliest Priority Filing Date: 02/02/2001

***For Sequence Searches Only* Please include all pertinent information (parent, child, divisional, or issued patent numbers) along with the appropriate serial number.**

Attached claims 1-7

Also Zr-alloy can be Zircaloy-2 or ~~Zirc~~ Zircaloy-4, Zr-2.5Nb, Reactor Grade Zirconium

****

STAFF USE ONLY		Type of Search		Vendors and cost where applicable	
Searcher:	Olee	NA Sequence (#)		STN	
Searcher Phone #:		AA Sequence (#)		Dialog	
Searcher Location:		Structure (#)		Questel/Orbit	
Date Searcher Picked Up:		Bibliographic		Dr.Link	
Date Completed:	7/12/02	Litigation		Lexis/Nexis	
Searcher Prep & Review Time:	3 hr	Fulltext		Sequence Systems	
Clerical Prep Time:		Patent Family		WWW/Internet	
Online Time:	4 hrs	Other		Other (specify)	

PTO-1590 (1-2000)

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=> d his

     (FILE 'HOME' ENTERED AT 10:47:43 ON 12 JUL 2002)

     FILE 'REGISTRY' ENTERED AT 10:47:54 ON 12 JUL 2002
L1          55532 S ZR/ELS AND AYS/CI
L2          53412 S ZR AND AYS/CI
L3          45373 S L1 AND 50-100/MAC

     FILE 'HCAPLUS' ENTERED AT 10:48:53 ON 12 JUL 2002
L4        1297849 S ALPHA?
L5          59969 S L4(3N)(PHASE? OR STRUCTUR? OR MICROSTRUCTUR?)
L6         178683 S MICROSTRUCTUR?
L7                QUE PRODUC? OR PROD# OR GENERAT? OR MANUF? OR MFR# OR CREAT? OR
L8        1084859 S CREEP? OR STRESS? OR STRAIN? OR DEFORMAT? OR FATIGUE? OR FRAC
L9         176174 S L8(4N)(RESIST? OR RECOVER? OR STRENGTH?) OR TOUGHNESS? OR RES
L10        737678 S NUCLEAR?
L11        193290 S L10(4N)FUEL? OR URANIUM? OR PLUTONIUM OR PU
L12          5494 S L10(3N)FUEL?(3N)CLAD?
L13               QUE CLAD? OR CASING? OR SHEATH? OR ENSHEATH? OR ENCAS? OR ENCAP
L14        695326 S LATH? OR STRIP? OR SWATH# OR BAND## OR SLAT### OR ROW###
L15        694636 S TUBE#  OR TUBING# OR TUBUL? OR TUBIFORM? OR PIPE# OR PIPING#
L16         27688 S (FAST## OR SWIFT## OR RAPID? OR QUICK?)(4N)(COOL?)
L17         25940 S (COLD# OR METAL?)(2N)(WORK? OR METALWORK?)
L18        250121 S ANNEAL? OR RECRYSTALLI?
L19         19467 S ZIRCALOY? OR (ZIRONI## OR ZR)(3N)(ALLOY? OR AMALGAM? OR MIXTU
L20         46961 S L3
L21         57035 S L19 OR L20 OR ZIRCALOY(2W)4
L22          7517 S ACICULAR
L23          4269 S NEEDLE?(3N)(LIKE#)
L24         11713 S L22 OR L23
L25               QUE HEAT? OR WARM? OR HOT# OR CALEFACT? OR TORREFACT? OR PYROL?

     FILE 'STNGUIDE' ENTERED AT 11:05:39 ON 12 JUL 2002

     FILE 'HCAPLUS' ENTERED AT 11:15:58 ON 12 JUL 2002
L26        946440 S 70/SC,SX OR 71/SC,SX
L27        599536 S 56/SX,SC
L28               QUE BETA
L29               QUE (BINARY OR DUAL OR TWO)(3N)PHASE?
L30         30738 S L21 AND L7
L31          2577 S L30 AND L4
L32          1144 S L30 AND L5
L34            66 S L32 AND (L14 OR LATH?)
L35            34 S L34 AND L8
L36             7 S L34 AND L9
L37           130 S L31 AND L14
L38            56 S L37 AND L8
L39            13 S L37 AND L9
L40             1 S L39 AND L10
L41          2802 S L21 AND L12
L42           292 S L41 AND L4
L43           120 S L41 AND L5
L44           190 S L42 AND L7
L45            71 S L43 AND L7
L46           190 S L44 AND L4
L47            71 S L45 AND L5
L48            71 S L44 AND L5
L49            71 S L45 AND L4
L50            80 S L46 AND L8
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L51          4 S L46 AND L9
L52         71 S L47 OR L48 OR L49
L53         38 S L52 AND L8
L54          4 S L52 AND L9
L55        293 S L41 AND L18
L56         54 S L55 AND L17
L57          2 S L56 AND L16
L58         60 S L55 AND L4
L59         39 S L58 AND L28
L60         31 S L59 AND L7
L61          1 S L60 AND L14
L62          2 S L60 AND L9
L63         15 S L36 OR L40 OR L51 OR L54 OR L57 OR L61 OR L62
L64          5 S L39 NOT L63

=> d cost
COST IN U.S. DOLLARS                           SINCE FILE     TOTAL
                                                    ENTRY   SESSION
CONNECT CHARGES                                     61.38    117.61
NETWORK CHARGES                                      1.86      4.74
SEARCH CHARGES                                       0.00     16.00
DISPLAY CHARGES                                     49.00     49.00
                                                  -------   -------
                                                   112.24    187.35
CAPLUS FEE (5%)                                      5.52      8.29
                                                  -------   -------
FULL ESTIMATED COST                                117.76    195.64

DISCOUNT AMOUNTS (FOR QUALIFYING ACCOUNTS)     SINCE FILE     TOTAL
                                                    ENTRY   SESSION
CA SUBSCRIBER PRICE                                -12.39    -12.39

IN FILE 'HCAPLUS' AT 11:34:34 ON 12 JUL 2002
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Harry,

I did a search in Chemical abstracts and another in Dialog.
I also included a Derwent record I got on EAST. By searching (nuclear$ near3 fuel$ near3 clad$) AND (zirconi$3 or zircaloy$3) I found a lot of good art (manuf. zircaloy cladding for nuclear applications).

I also included Derwent record so you look at Manual codes (CPI codes). K05-B04B and M26-B06C are derwent codes. The are derwents classification system. K05 is for nuclear reactors and M26 is for nonferrous alloys. To search with codes, just end with .cpi.,for example M26-B06C.cpi. And of course you can truncate as well.

It is very powerful, and you can pull up good art very quickly.
If you have any questions, feel free to call anytime.

John

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? show file

File   6:NTIS 1964-2002/Jul W3
         (c) 2002 NTIS, Intl Cpyrght All Rights Res
File   2:INSPEC 1969-2002/Jul W1
         (c) 2002 Institution of Electrical Engineers
File   8:Ei Compendex(R) 1970-2002/Jul W1
         (c) 2002 Engineering Info. Inc.
File  62:SPIN(R) 1975-2002/Jun W3
         (c) 2002 American Institute of Physics
File  65:Inside Conferences 1993-2002/Jul W1
         (c) 2002 BLDSC all rts. reserv.
File  77:Conference Papers Index 1973-2002/May
         (c) 2002 Cambridge Sci Abs
File  94:JICST-EPlus 1985-2002/May W3
         (c)2002 Japan Science and Tech Corp(JST)
File 103:Energy SciTec 1974-2002/Jun B2
         (c) 2002 Contains copyrighted material
File 109:Nuclear Sci. Abs. 1948-1976
         (c)1997 Contains copyrighted material
File 347:JAPIO Oct 1976-2002/Mar(Updated 020702)
         (c) 2002 JPO & JAPIO
File 351:Derwent WPI 1963-2002/UD,UM &UP=200244
         (c) 2002  Thomson Derwent
File  32:METADEX(R) 1966-2002/Aug B1
         (c) 2002 Cambridge Scientific Abs
? ds

Set     Items   Description
S1       3306   NUCLEAR?(3N)CLAD?
S2       6779   NUCLEAR?(3N)(CLAD? OR CASING? OR SHEATH? OR ENSHEATH? OR E-
             NCAS? OR ENCAPSULAT? OR ENVELOP? OR OVERLAID? OR LAMIN? OR LA-
             MEL? OR ENCAS? OR WRAP? OR SURROUND?)
S3    1957570   NUCLEAR?
S4      65309   S3 AND (CLAD? OR CASING? OR SHEATH? OR ENSHEATH? OR ENCAS?
             OR ENCAPSULAT? OR ENVELOP? OR OVERLAID? OR LAMIN? OR LAMEL? OR
              ENCAS? OR WRAP? OR SURROUND?)
S5      87445   ZIRCALOY? OR (ZIRCONI? OR ZR)(4N)(ALLOY? OR AMALGAM?)
S6      63960   ALPHA(3N)(PHASE? OR STRUCTUR? OR MICROSTRUCTUR?)
S7      40457   S5 AND (PRODUC? OR PROD? ? OR GENERAT? OR MANUF? OR MFR? ?
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              OR CREAT? OR FORMING? ? OR FORMAT? OR MAKE? ? OR MADE? ? OR M-
              AKING? ? OR FABRICAT? OR PREPAR? OR PREP? ?)
S8    3615733   TUBE? ? OR TUBING? ? OR TUBUL? OR TUBIFORM? OR TUBELIKE? OR
               PIPE? ? OR PIPING? ? OR PIPELI? OR CONDUIT? OR CYLIND?
S9    1849880   LATH? OR STRIP? OR SWATH? OR BAND? OR SLAT?
S10     10038   ACICULAR? OR NEEDLE?(3N)LIKE?
S11     55548   (COARSE? OR LARGE? OR BIG?)(3N)(GRAIN?)
S12    174165   CREEP?
S13     16550   S12(4N)(RESIST?)
S14      3962   NUCLEAR?(5N)CLAD?
S15     24564   NUCLEAR? AND CLAD?
S16    122622   S3 AND S8
S17      7023   S7 AND S8
S18      2474   S17 AND CLAD?
S19      8723   S5 AND (PRODUC? OR PROD? ? OR GENERAT? OR MANUF? OR MFR? ?
              OR CREAT? OR FORMING? ? OR FORMAT? OR MAKE? ? OR MADE? ? OR M-
              AKING? ? OR FABRICAT? OR PREPAR? OR PREP? ?)/TI
S20      1625   S19 AND S3
S21       207   S19 AND S14
S22      5423   S5 AND (PRODUC? OR PROD? ? OR MANUF? OR MFR? ? OR FABRICAT?
               OR PREPAR? OR PREP? ?)/TI
S23       176   S22 AND S14
S24        27   S23 AND ALPHA?
S25       137   S23 AND S8
S26         6   S25 AND S12
S27         2   S23 AND S9
S28         0   S25 AND LATH?
S29         0   S23 AND S10
S30        20   S23 AND GRAIN?
S31         3   S23 AND S11
S32        27   S23 AND COLD?(4N)WORK?
S33         8   S26 OR S31
S34        42   (S32 OR S24) NOT S33
S35        40   RD S34 (unique items)
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? t s33/7,de/1-8

33/7,DE/1 (Item 1 from file: 6)
DIALOG(R)File 6:NTIS

1191935 NTIS Accession Number: DE85011649
Manufacturing Process to Reduce Large Grain Growth in Zirconium Alloys (Patent Application)
Rosecrans, P. M.
Department of Energy, Washington, DC.
Corp. Source Codes: 052661000
Report No.: PAT-APPL-6-636 659
Filed 1 Aug 84 11p
Languages: English Document Type: Patent
Journal Announcement: GRAI8521; NSA1000

This Government-owned invention available for U.S. licensing and, possibly, for foreign licensing. Copy of application available NTIS. Portions of this document are illegible in microfiche products.

NTIS Prices: PC A02/MF A01
Country of Publication: United States
Contract No.: AC12-76SN00052

It is an object of the present invention to provide a procedure for desensitizing zirconium-based alloys to large grain growth (LGG) during

thermal treatment above the recrystallization temperature of the alloy. It is a further object of the present invention to provide a method for treating zirconium-based alloys which have been cold-worked in the range of 2 to 8% strain to reduce large grain growth. It is another object of the present invention to provide a method for fabricating a zirconium alloy clad nuclear fuel element wherein the zirconium clad is resistant to large grain growth. (ERA citation 10:030822)
Descriptors: *Reactor Materials; *Zirconium Base Alloys; Fabrication; Grain Growth; Heat Treatments; Metallurgy; Microstructure

33/7,DE/2 (Item 1 from file: 8)
DIALOG(R)File 8:Ei Compendex(R)

05491054
E.I. No: EIP00035071486
Title: Influence of composition and fabrication process on out-of-pile and in-pile properties of M5 alloy
Author: Mardon, Jean-Paul; Charquet, Daniel; Senevat, Jean
Corporate Source: FRAMATOME Nuclear Fuel, Lyon, Fr
Conference Title: 12th ASTM International Symposium: Zirconium in the Nuclear Industry
Conference Location: Toronto, Que, Can Conference Date: 19980115-19980118
E.I. Conference No.: 56408
Source: ASTM Special Technical Publication n 1354 2000. p 505-524
Publication Year: 2000
CODEN: ASTTA8 ISSN: 1040-3094
Language: English
Document Type: JA; (Journal Article) Treatment: X; (Experimental)
Journal Announcement: 0004W3
Abstract: Within the scope of the optimization of the M5 cladding tubes made of ternary alloy (Zr, Nb, O), an extensive program of investigation and industrial development has been undertaken. The various possible factors and potential causes of variability have been thoroughly analyzed with the aid of industrial-scale or laboratory ingots from the point of view of their impact on the finished product properties. In this way, all the chemical composition variabilities of the alloying elements (Nb, O) and impurities (Fe, S, C) have been studied through variable-composition ingots. Also, a number of manufacturing process variants (number of melts, quench, extrusion, heat treatments, pilgering . . .) have been studied. In some cases, it was possible to investigate the combined effect of two types of parameters (sulfur-process and iron-process interactions). For each of the products manufactured in this way, systematic, characterization of: creep, microstructure (optical microscopy, TEM), corrosion (autoclave) tests was accomplished. In each case, the influence of each variability parameter was tested, and in many cases correlations with the out-of-pile characteristics of the finished tubes were established. Lastly, for some variables (process, S content, . . .) the effect of irradiation was more specifically analyzed. These investigations pointed to a new and very important factor, the effect of sulfur concentration on the in-pile operating properties, especially creep and growth. This set of results constitutes a database covering the whole industrial variability range of this alloy, allowing Framatome to embark upon its industrial development phase and to offer M5 cladding tube on the market. This product has been irradiated over a wide range of PWR service and environmental conditions in Europe and the U.S. The improvements in corrosion (Factors 3 to 4), hydriding (Factors 5 to 6), and creep and growth (Factors 2 to 3) data after five cycles (55 GWd/tU) show impressive gains over optimized low-tin

Zircaloy-4. (Author abstract) 12 Refs.
Descriptors: *Zirconium alloys; Nuclear fuel cladding; Alloying elements; Niobium; Oxygen; Iron; Sulfur; Carbon; Crystal impurities; Ingots

33/7,DE/3 (Item 1 from file: 103)
DIALOG(R)File 103:Energy SciTec

01738261 AIX-17-019614; EDB-86-061943
Author(s): Nakajima, Junjiro; Umehara, Hajime; Inagaki, Masatoshi
Title: Nuclear fuel cladding tubes and its fabrication
Corporate Source: Hitachi Ltd., Tokyo (Japan)
Patent No.: JP 60-36984 A
Patent Assignee(s): Hitachi Ltd., Tokyo, Japan
Patent Date Filed: Filed date 9 Aug 1983
Publication Date: 26 Feb 1985
p 6
Note: JP patent application 58-145860
Language: Japanese
Availability: JAPIO. Also available from INPADOC.
Abstract: The purpose of this patent is to improve the corrosion-resistance, stress corrosion-resistance and high temperature creep properties of fuel cladding tubes. In a fuel cladding tube having a fuel cladding layer made of a zirconium-based alloy, the outer surface layer, the inner surface layer and the intermediate layer between the outer and the inner surface layers is made of substantially complete recrystallization structure. The grain size is made larger in the order of the outer surface layer, intermediate layer and the inner surface layer.
Major Descriptors: *FUEL CANS -- FABRICATION; *FUEL CANS -- GRAIN SIZE
Descriptors: CORROSION RESISTANCE; CREEP; RECRYSTALLIZATION; STRESS CORROSION; ZIRCONIUM BASE ALLOYS
Broader Terms: ALLOYS; CHEMICAL REACTIONS; CORROSION; CRYSTAL STRUCTURE; MECHANICAL PROPERTIES; MICROSTRUCTURE; SIZE; ZIRCONIUM ALLOYS

33/7,DE/4 (Item 2 from file: 103)
DIALOG(R)File 103:Energy SciTec

01587381 EDB-85-094160
Author(s): Rosecrans, P.M.
Title: Manufacturing process to reduce large grain growth in zirconium alloys
Corporate Source: General Electric Co., Schenectady, NY (USA)
Patent Assignee(s): Dept. of Energy
Application/Priority No.: US 6-636659
Publication Date: 1 Aug 1984
p 11
Order Number: DE85011649
Contract Number (DOE): AC12-76SN00052
Note: Portions of this document are illegible in microfiche products
Language: English
Availability: NTIS, PC A02/MF A01; 1.
Abstract: It is an object of the present invention to provide a procedure for desensitizing zirconium-based alloys to large grain growth (LGG) during thermal treatment above the recrystallization temperature of the alloy. It is a further object of the present invention to provide a method for treating zirconium-based alloys which have been cold-worked

in the range of 2 to 8% strain to reduce large grain growth. It is another object of the present invention to provide a method for fabricating a zirconium alloy clad nuclear fuel element wherein the zirconium clad is resistant to large grain growth.
Major Descriptors: *REACTOR MATERIALS -- FABRICATION; *ZIRCONIUM BASE ALLOYS -- FABRICATION; *ZIRCONIUM BASE ALLOYS -- METALLURGY
Descriptors: GRAIN GROWTH; HEAT TREATMENTS; MICROSTRUCTURE
Broader Terms: ALLOYS; CRYSTAL STRUCTURE; MATERIALS; ZIRCONIUM ALLOYS

33/7,DE/5 (Item 1 from file: 347)
DIALOG(R)File 347:JAPIO

04249278
PRODUCTION METHOD FOR FUEL CLADDING

PUB. NO.: 05-240978 [JP 5240978 A] ,
PUBLISHED: September 21, 1993 (19930921)
INVENTOR(s): NAKAJIMA JUNJIRO
UMEHARA HAJIME
INAGAKI MASATOSHI
APPLICANT(s): HITACHI LTD [000510] (A Japanese Company or Corporation), JP (Japan)
APPL. NO.: 03-338154 [JP 91338154]
FILED: December 20, 1991 (19911220)
JAPIO CLASS: 23.1 (ATOMIC POWER -- General)

ABSTRACT
PURPOSE: To obtain a production method for fuel cladding for nuclear rectors superior in anti-corrosion, anti-stress corrosion and high temperature creep characteristic.

CONSTITUTION: To a pushed complex tube 22 consisting of a complex pilet assembled by a zirconium base alloy hollow pilet as an outer tube and a zirconium hollow pilet as an inner tube, sealed at both ends and treated in heat, a melt processing is applied with high frequency hardening using a high frequency induction heating coil 26 and a cooling nozzle 27 which move in the axial direction relatively to the outer surface of the pushed complex tube 22 in the state coolant is circulated inside. Then for repeating cold rolling and annealing by turns, melt processing is performed by supporting the upper part and the lower part of the pushed complex tube 22 with an upper support 23 and a lower support 24 with the same material as the complex pilet.

33/7,DE/6 (Item 1 from file: 351)
DIALOG(R)File 351:Derwent WPI

012460096
WPI Acc No: 1999-266204/199923
Zirconium alloy nuclear fuel cladding production
Patent Assignee: MITSUBISHI MATERIALS CORP (MITV)
Inventor: ISOBE T; SUDA Y
Number of Countries: 003 Number of Patents: 003
Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week	
FR 2769637	A1	19990416	FR 9812784	A	19981013	199923	B
JP 11194189	A	19990721	JP 98287800	A	19981009	199939	

US 6125161	A	20000926	US 98169968	A	19981013	200051
			US 99397094	A	19990916	

Priority Applications (No Type Date): JP 98287800 A 19981009; JP 97278935 A 19971013
Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
FR 2769637	A1		39	C21D-008/00	
JP 11194189	A		24	G21C-003/06	
US 6125161	A			G21C-003/07	Div ex application US 98169968

Abstract (Basic): FR 2769637 A1
Abstract (Basic):
NOVELTY - In the production of nuclear fuel cladding of a zirconium alloy containing Nb or Nb+Ta, annealing is carried out at 550-850degreesC for 1-4 h such that the log of the cumulative anneal parameter is -20 to -15 and satisfies a mathematical relationship relating it to the Nb or Nb+Ta content.

DETAILED DESCRIPTION - Nuclear fuel cladding is produced by subjecting a zirconium alloy of composition (by wt.) 0.2-1.7% Sn, 0.18-0.6% Fe, 0.07-0.4% Cr, 0.05-1.0% Nb, optionally 0.01-0.1% Ta, balance zirconium and impurities, including at most60 ppm N, to hot forging, solution heat treatment, hot extrusion, repeated annealing and cold rolling, and final stress relief annealing, the annealing being carried out at 550-850degreesC for 1-4 h such that the cumulative anneal parameter approximatelySAi (where approximatelySAi=approximatelyStiasteriskexp(-40000/Ti)) satisfies the relationships of logapproximatelySAi=-20 to -15 and logapproximatelySAi=-18-10XNb to -15-3.75(XNb-0.2), in which Ai=anneal parameter for the 'i'th anneal, ti=anneal duration (h) for the 'i'th anneal, Ti=the anneal temperature (K) for the 'i'th anneal and XNb=the Nb and optional Ta content (in wt.%). An INDEPENDENT CLAIM is also included for a zirconium alloy nuclear fuel cladding made by the above process.

USE - For producing nuclear fuel cladding tubes for a PWR.

ADVANTAGE - The annealing conditions provide the cladding tube with better corrosion resistance and creep properties than conventional cladding tubes and thus has a long useful life.

pp; 39 DwgNo 0/0
Title Terms: ZIRCONIUM; ALLOY; NUCLEAR; FUEL; CLAD; PRODUCE
Derwent Class: K05; M26; M29; X14
International Patent Class (Main): C21D-008/00; G21C-003/06; G21C-003/07
International Patent Class (Additional): C21D-001/26; C22C-016/00; C22F-001/00; C22F-001/18

33/7,DE/7 (Item 2 from file: 351)
DIALOG(R)File 351:Derwent WPI

010561475
WPI Acc No: 1996-058429/199606
Mfr. of of Zr alloys tubes for nuclear reactor fuel - by extruding, cold rolling and quenching achieving low irradiation induced axial growth with high transversal creep strength and good corrosion resistance during irradiation
Patent Assignee: SANDVIK AB (SANV)
Inventor: ANDERSSON T; ANDERSON T
Number of Countries: 019 Number of Patents: 010
Patent Family:
Patent No Kind Date Applicat No Kind Date Week

WO 9535395	A1	19951228	WO 95SE749	A	19950620	199606	B
SE 9402250	A	19960129	SE 942250	A	19940622	199615	
EP 760017	A1	19970305	EP 95923640	A	19950620	199714	
			WO 95SE749	A	19950620		
JP 10501846	W	19980217	WO 95SE749	A	19950620	199817	
			JP 96502068	A	19950620		
KR 97704064	A	19970809	WO 95SE749	A	19950620	199836	
			KR 96707338	A	19961221		
US 5876524	A	19990302	WO 95SE749	A	19950620	199916	
			US 97765590	A	19970417		
EP 760017	B1	19990908	EP 95923640	A	19950620	199941	
			WO 95SE749	A	19950620		
DE 69512052	E	19991014	DE 612052	A	19950620	199949	
			EP 95923640	A	19950620		
			WO 95SE749	A	19950620		
ES 2135749	T3	19991101	EP 95923640	A	19950620	199953	
SE 513488	C2	20000918	SE 942250	A	19940622	200054	

Priority Applications (No Type Date): SE 942250 A 19940622
Cited Patents: 03Jnl.Ref; EP 488027; US 2894866; US 4450020; US 5266131; WO 9501639
Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
WO 9535395	A1	E	18	C22C-016/00	
Designated States (National): JP KR US					
Designated States (Regional): AT BE CH DE DK ES FR GB GR IE IT LU MC NL PT SE					
SE 9402250	A			C22F-001/18	
EP 760017	A1	E		C22C-016/00	Based on patent WO 9535395
Designated States (Regional): DE ES FR GB IT SE					
JP 10501846	W		15	C22F-001/18	Based on patent WO 9535395
KR 97704064	A			C22C-016/00	Based on patent WO 9535395
US 5876524	A			C22F-001/18	Based on patent WO 9535395
EP 760017	B1	E		C22C-016/00	Based on patent WO 9535395
Designated States (Regional): DE ES FR GB IT SE					
DE 69512052	E			C22C-016/00	Based on patent EP 760017
					Based on patent WO 9535395
ES 2135749	T3			C22C-016/00	Based on patent EP 760017
SE 513488	C2			C22F-001/18	

Abstract (Basic): WO 9535395 A

Mfr. of cladding tubes and structural tube parts for fuel of Zr alloys for application in nuclear reactors is claimed. The Zr alloy is initially extruded and then followed by a series of cold rolling passes accompanied with alpha-annealing steps. beta-quenching is then performed by heating in the beta-phase range (950-1250 deg. C) until the structure is 100% beta-phase, then quenching at 100-450 deg. C s-1, transforming the entire tube to alpha-phase followed by a vacuum anneal in the alpha-phase for suitable time and temperature to produce an annealing parameter of 3.4x10-16x10-13.

USE - The method of manufacture relates to the production of cladding for nuclear core materials and structural parts in fuel element skeletons, using Zr based alloys.

ADVANTAGE - Compared to prior art material, this invention provides lower irradiation induced axial growth with higher transversal creep strength and good corrosion resistance during irradiation. Tests for corrosion resistance conducted in steam at 400 deg. C for 60 days the invention showed a weight gain of 2-4 mg dm-3 less than prior art. Creep tests at 400 deg. C for 240 hours with a peripheral tension of 130 MPa. Transversal creep elongation of 0.45-0.70% was found in this

invention compared to 1.8-2.0% in the prior art. Axial growth was compared by the use of Kearns factor, fa. Axial growth is found to be greater in material with a lower fa factor. The invention has a fa value in the range 0.26-0.32 while conventional tubes have 0.03-0.07.
Dwg.0/0
Title Terms: MANUFACTURE; ALLOY; TUBE; NUCLEAR; REACTOR; FUEL; EXTRUDE; COLD; ROLL; QUENCH; ACHIEVE; LOW; IRRADIATE; INDUCE; AXIS; GROWTH; HIGH; TRANSVERSE; CREEP; STRENGTH; CORROSION; RESISTANCE; IRRADIATE
Derwent Class: K05; M26; M29; X14
International Patent Class (Main): C22C-016/00; C22F-001/18
International Patent Class (Additional): C22F-001/00; G21C-003/06

33/7,DE/8 (Item 3 from file: 351)
DIALOG(R)File 351:Derwent WPI

007736305
WPI Acc No: 1989-001417/198901
Zirconium alloy nuclear fuel cladding mfr. - involving final beta phase heat treatment for improved corrosion and creep resistance
Patent Assignee: COMMISSARIAT ENERGIE ATOMIQUE (COMS); FRAMATOME SA (FRAT); URANIUM PECHINEY & FRAMATOME ZIRCOTUBE (UGIN); FRAMATOME (FRAT); SNC URANIUM PECH & FRAMA (UGIN); SNC URANIUM PECH & FRAMAT (UGIN)
Inventor: DECOURS J; MARDON J P; PELCHAT J; WEISZ M; LEPAPE J; LE PAPE J
Number of Countries: 011 Number of Patents: 008
Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week	
EP 296972	A	19881228	EP 88401573	A	19880622	198901	B
JP 1097897	A	19890417	JP 88153689	A	19880623	198921	
CN 1030261	A	19890111				198949	
ZA 8804447	A	19900228	ZA 884447	A	19880622	199013	
US 4938921	A	19900703	US 88210444	A	19880623	199029	
EP 296972	B1	19920812	EP 88401573	A	19880622	199233	
DE 3873643	G	19920917	DE 3873643	A	19880622	199239	
			EP 88401573	A	19880622		
ES 2034312	T3	19930401	EP 88401573	A	19880622	199323	

Priority Applications (No Type Date): FR 878814 A 19870623
Cited Patents: DE 2008320; DE 2651870; DE 2951096; DE 2951102; EP 213771; EP 98996; FR 2368547; US 4238251; US 423851
Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
EP 296972	A	F	8		
Designated States (Regional): BE DE ES FR GB IT SE					
EP 296972	B1	F		C22F-001/18	
Designated States (Regional): BE DE ES FR GB IT SE					
DE 3873643	G			C22F-001/18	Based on patent EP 296972
ES 2034312	T3			C22F-001/18	Based on patent EP 296972

Abstract (Basic): EP 296972 A

In the mfr. of zirconium alloy tubing for fuel cladding, using several successive rolling passes and anneals, the novelty is that the final stage is a beta phase homogenisation heat treatment at 950-1250 deg.C, followed by rapid cooling to ambient temp.

USE/ADVANTAGE - The process is used in the mfr. of PWR nuclear fuel cladding, has improved general corrosion resistance under irradiation at its external surface, improved internal corrosion resistance under stress and irradiation, improved radial creep resistance at high temps. and under neutron flux, improved axial creep resistance and reduced axial extension. The fuel elements can remain longer in the reactor

core since the cladding remains sealed and has long term resistance to irradiation.
0/0
Abstract (Equivalent): EP 296972 B
In the mfr. of zirconium alloy tubing for fuel cladding, using several successive rolling passes and anneals, the novelty is that the final stage is a beta phase homogenisation heat treatment at 950-1250 deg.C followed by rapid cooling to ambient temp.
USE/ADVANTAGE - The process is used in the mfr. of PWR nuclear fuel cladding , has improved general corrosion resistance under irradiation at its external surface, improved internal corrosion-resistance under stress and irradiation, improved radial creep resistance at high temps. and under neutron flux, improved axial creep resistance and reduced axial extension. The fuel elements can remain longer in the reactor core since the cladding remains sealed and has long term resistance to irradiation.a
Abstract (Equivalent): US 4938921 A
Zr alloy tube for a fuel element sheath in a nuclear reactor is mfd. from Zircalloy-4 alloy contg. (%) 1.2-1.7Sn, 0.18-0.24 Fe, 0.7-0.13 Cr, such that fe-Cr= 0.28 min., with 80-270 ppm C and 900-1600 ppm O2. The tube is produced in a successium of cold rolling and annealing steps, including a final beta phase heat treatment in which the tube is maintained at 950-1250deg.C for a time sufficient to obtain a homogeneous beta phase through the wall thickness, followed by rapid cooling to ambient emp. to retain the beta phase.
ADVANTAGE - Enhanced corrosion resistance and creep resistance.
Title Terms: ZIRCONIUM; ALLOY; NUCLEAR; FUEL; CLAD; MANUFACTURE; FINAL; BETA; PHASE; HEAT; TREAT; IMPROVE; CORROSION; CREEP; RESISTANCE
Index Terms/Additional Words: ZIRCONIUM
Derwent Class: K05; M26
International Patent Class (Main): C22F-001/18
International Patent Class (Additional): C22C-016/00; G21C-003/06

? t s35/7,de/1-40

35/7,DE/1 (Item 1 from file: 2)
DIALOG(R)File 2:INSPEC

5069574 INSPEC Abstract Number: A9521-8160B-035
Title: Remote Raman spectroscopic studies of corrosion products formed on nuclear fuel claddings used in PWR and AGR systems
Author(s): Edwards, H.G.M.; Long, D.A.; Willis, I.T.
Author Affiliation: Dept. of Chem. & Chem. Technol., Bradford Univ., UK
Journal: Journal of Raman Spectroscopy vol.26, no.8-9 p.757-62
Publication Date: Aug.-Sept. 1995 Country of Publication: UK
CODEN: JRSPAF ISSN: 0377-0486
U.S. Copyright Clearance Center Code: 0377-0486/95/080757-06
Language: English Document Type: Journal Paper (JP)
Treatment: Experimental (X)
Abstract: The construction of a remote Raman microscope system with a variable microscope-to-spectrometer distance of up to 3 m is described. Oxidative surface corrosion products on zirconium alloy and stainless-steel nuclear fuel claddings for pressurized water and advanced gas cooled reactor systems produced under various conditions were studied and the potential of the technique for in situ analysis is discussed. Under the conditions studied experimentally, the major corrosion products are found to be alpha -ZrO/sub 2/ and alpha -Fe/sub 2/O/sub 3/ for the zirconium alloy and stainless-steel claddings, respectively. (29 Refs)

Subfile: A
Descriptors: corrosion; fission reactor fuel claddings; oxidation; Raman spectra; stainless steel; zirconium alloys

35/7,DE/2 (Item 1 from file: 8)
DIALOG(R)File 8:Ei Compendex(R)

05491051
E.I. No: EIP00035071483
Title: New fabrication process for Zr-lined Zircaloy-2 tubing
Author: Abe, Hideaki; Takeda, Kiyoko; Uehira, Akihiro; Anada, Hiroyuki; Furugen, Munekatsu
Corporate Source: Sumitomo Metal Industries Ltd, Hyogo, Jpn
Conference Title: 12th ASTM International Symposium: Zirconium in the Nuclear Industry
Conference Location: Toronto, Que, Can Conference Date: 19980115-19980118
E.I. Conference No.: 56408
Source: ASTM Special Technical Publication n 1354 2000. p 425-459
Publication Year: 2000
CODEN: ASTTA8 ISSN: 1040-3094
Language: English
Document Type: JA; (Journal Article) Treatment: T; (Theoretical)
Journal Announcement: 0004W3

Abstract: A new fabrication process for Zr-lined Zircaloy-2 cladding tubes was developed, including a cold pilgering pass schedule and an appropriate heat treatment. In this study, the effect of tool design in cold pilgering on the quality of tubes was investigated. Cold pilgering tests were performed using tools with different tool curves. Simultaneously, for each tool the plastic strain and stress in the tubes during cold pilgering were simulated using a theoretical plastic deformation model. The investigations indicated that an abrupt change in the strain and stress in the tubes during cold pilgering should be avoided to prevent crack formation in the tubes. The results made possible the use of higher reduction (91%) in cold pilgering. Also, an appropriate heat treatment for the properties of the final tubes in the new process was investigated. It indicated that quenching the tubeshell and the intermediate annealing temperature had a small effect on the mechanical properties. To obtain both nodular and uniform corrosion resistance the appropriate precipitate size was in the range 140 to 170 nm. beta -quenching of the tubeshell was not effective for improving the uniform corrosion resistance. An appropriate intermediate annealing temperature was in the range 823 to 953 K for the alpha plus beta quenched and for the alpha -annealed tubeshells. These results led to the new fabrication process for Zr-lined Zircaloy-2 cladding tubes (12 mm outside diameter) in which the final tube was fabricated using only two cold pilgering passes from the tubeshell (63.5 mm outside diameter). (Author abstract) 19 Refs.

Descriptors: *Zirconium alloys; Nuclear fuel cladding; Tubes (components) ; Cold rolling; Strain; Stress analysis; Plastic deformation; Quenching; Annealing; Corrosion resistance

35/7,DE/3 (Item 2 from file: 8)
DIALOG(R)File 8:Ei Compendex(R)

04213546

E.I. No: EIP95072798287
Title: Influence of manufacturing process on the in-reactor creep anisotropy of stress-relieved Zircaloy-2 cladding
Author: Shann, S.H.; Van Swam, L.F.
Corporate Source: Siemens Power Corp, Richland, WA, USA
Source: Nuclear Engineering and Design v 156 n 3 Jun 2 1995. p 351-358
Publication Year: 1995
CODEN: NEDEAU ISSN: 0029-5493
Language: English
Document Type: JA; (Journal Article) Treatment: T; (Theoretical); X; (Experimental)
Journal Announcement: 9509W4
Abstract: A procedure to determine the axial/radial and circumferential/radial contractile strain ratios (the R and P factors respectively in the Backofen-modified von Mises-Hill yield criterion) from post-irradiation dimensional measurements of Zircaloy-2 cladding of BWR fuel rods, tie rods and water rods was developed and has been described previously (S.H. Shann and L.F. van Swam, Creep anisotropy of Zircaloy-2 cladding during irradiation, Trans. SMiRT-11, Vol. C, 1991). The present study employs the procedure to determine the anisotropy factors R and P for textured cold-worked stress-relieved (CWSR) Zircaloy-2 cladding fabricated by various manufacturing processes. The analysis indicates that the cladding manufacturing process can have a pronounced effect on the anisotropy of irradiation-induced creep. Cladding types with identical yield and ultimate tensile strengths but fabricated by different manufacturing processes have different values of R and P during in-reactor creep. (Author abstract) 4 Refs.
Descriptors: *Nuclear fuel cladding; Anisotropy; Zirconium alloys; Irradiation; Stresses; Processing; Creep; Strain; Mathematical models; Tensile strength

35/7,DE/4 (Item 3 from file: 8)
DIALOG(R)File 8:Ei Compendex(R)

00189643
E.I. Monthly No: EI71X177920
Title: Effects of fabrication conditions on mechanical properties of zircaloy tubes for nuclear fuel cladding.
Author: KONISHI, T.; MATSUDA, K.
Source: Sumitomo Metals v 23 n 1 Jan 1971 p 40-8
Publication Year: 1971
CODEN: SUMMA
Language: JAPANESE
Journal Announcement: 71X1
Abstract: Mechanical properties and related features of zircaloy tubes are positively adjustable in wide range by controlling the fabrication conditions. In general, choice of the reducing process which would give large final reduction, large final Q value and large total Q value in cold working stage will yield versatile material that can meet varieties of specifications by adjusting subsequent finishing processes. RT value will be newly defined in the report. This value (analogous to r value used in deep drawing of sheet) is considered to be an effective measure in discussing the mechanical properties and related features of zircaloy tubes. In Japanese with English synopsis.
Descriptors: *TUBES--*Zircaloy; TUBES MANUFACTURE

35/7,DE/5 (Item 1 from file: 94)

DIALOG(R)File 94:JICST-EPlus

00380277 JICST ACCESSION NUMBER: 87A0107292 FILE SEGMENT: JICST-E
Manufacture and inspection techniques of Zircaloy nuclear fuel cladding tubes, and product quality.
TAKAISHI KAZUHIDE (1); MIYAJI MASATOSHI (1); WAKAMATSU RYUJI (1)
(1) Kobeseikosho Chofukitakojo
R & D / Kobe Seiko Giho(Kobe Steel Engineering Reports), 1987, VOL.37,NO.1, PAGE.5-9, FIG.9, TBL.2
JOURNAL NUMBER: F0164ABB ISSN NO: 0373-8868
UNIVERSAL DECIMAL CLASSIFICATION: 621.771.2/.8 621.039.54
LANGUAGE: Japanese COUNTRY OF PUBLICATION: Japan
DOCUMENT TYPE: Journal
ARTICLE TYPE: Commentary
MEDIA TYPE: Printed Publication
ABSTRACT: As the result of extensive studies on fabrication techniques of nuclear fuel cladding tubes since 1957, the comprehensive worldwide supply of high-quality Zircaloy-2 fuel cladding tubes for BWRs has been established and annual production capacity has now reached 650000meters. Recently, a fabrication technique of zirconium-lined Zircaloy-2 nuclear fuel cladding tubes has been developed, and qualification by customers was finished in 1986.(author abst.)
DESCRIPTORS: light water reactor; fuel can; mass production; tube rolling; cold rolling; degreasing; annealing; polishing(machining); acid cleaning; hardening(heat treatment); Zircaloy; zirconium; clad material ; lining; corrosion resistance; local corrosion; blast cleaning; sand particle
BROADER DESCRIPTORS: thermal neutron reactor; nuclear reactor; fuel element ; reactor component; production method; method; rolling(plastic working); plastic working; working and processing; tubemaking; manufacturing; cold working; removal; heat treatment; treatment; machining; chemical cleaning; cleaning(washing); cleaning(purification) ; zirconium base alloy; nonferrous alloy; alloy; metallic material; 4A group element; transition metal; metallic element; element; material; operation(processing); resistance(endure); corrosion; sand; clastic sediment; sediment; soil; particle

35/7,DE/6 (Item 1 from file: 103)
DIALOG(R)File 103:Energy SciTec

04163735 JPN-97-004434; EDB-97-072439
Title: Nuclear fuel cladding tube and method of manufacturing the same
Author(s)/Editor(s): Higashinakagawa, Emiko; Kubo, Hiroshi; Obata, Minoru; Hisatsune, Yoshimi.
Corporate Source: Toshiba Corp., Kawasaki, Kanagawa (Japan)
Patent No.: JP 9-5485 A
Patent Assignee(s): Toshiba Corp., Kawasaki, Kanagawa (Japan)
Priority No.: JP 7-155662
Patent Date Filed: 22 Jun 1995
Publication Date: 10 Jan 1997
(6 p)
Language: Japanese
Availability: Available from JAPIO. Also available from EPO
Abstract: A cladding tube main body made of a zirconium alloy and an end plug are joined by welding. Tensile stresses at the weld heat-affected portion between the cladding tube main body and the end plug are removed, so that compression stresses of 0 MPa or more but less than

the endurance strength of the zirconium alloy is applied on the weld heat affected portion. As the zirconium alloy, a zircaloy-2 or zircaloy-4 is preferable since it is excellent in the corrosion resistance and strength. The zirconium alloy may preferably be used also to the material of the end plug. The treatment for the removal of the tensile stresses includes a method of applying annealing to the weld heat-affected portion or a method of applying compression stresses thereto by applying external force such as a shot peening treatment. This can suppress occurrence of nodular corrosion and white homogeneous corrosion caused in the vicinity of the welded portion. (I.N.)
Major Descriptors: *FUEL CANS -- WELDING
Descriptors: ANNEALING; CLOSURES; COMPRESSION; HEAT AFFECTED ZONE; SHOT PEENING; STRESSES; TENSILE PROPERTIES; ZIRCONIUM ALLOYS
Broader Terms: ALLOYS; COLD WORKING; FABRICATION; HEAT TREATMENTS; JOINING; MATERIALS WORKING; MECHANICAL PROPERTIES; SURFACE TREATMENTS; ZONES

35/7,DE/7 (Item 2 from file: 103)
DIALOG(R)File 103:Energy SciTec

03891248 JPN-95-008503; EDB-95-135016
Title: Production process for nuclear fuel cladding tube
Author(s)/Editor(s): Hisatsune, Yoshimi; Higashinakagawa, Emiko; Arai, Shinji; Ikeda, Tadahiro.
Corporate Source: Toshiba Corp., Kawasaki, Kanagawa (Japan)
Patent No.: JP 7-109554 A
Patent Assignee(s): Toshiba Corp., Kawasaki, Kanagawa (Japan)
Priority No.: JP 5-276000
Patent Date Filed: 8 Oct 1993
Publication Date: 25 Apr 1995
(6 p)
Language: Japanese
Availability: Available from JAPIO. Also available from EPO.
Abstract: A Zr liner layer is formed on the inner surface of an external tube comprising a Zr alloy, then the external tube and the Zr liner layer are rapidly heated to a high temperature of [alpha] region kept for a short period of time, and then immediately quenched. With such procedures, there can be attained a long-life nuclear fuel cladding tube with excellent uniform corrosion resistance also in the liner portion on the inner side of the Zr alloy tube, and less degradation even upon long time use in a reactor atmosphere. (T.M.).
Major Descriptors: *CORROSION RESISTANCE -- QUENCHING; *CORROSION RESISTANCE -- TEMPERATURE DEPENDENCE; *NUCLEAR FUELS -- FUEL CANS; *ZIRCONIUM ALLOYS -- NODULAR CORROSION
Descriptors: ENVIRONMENT; LIFETIME; PIPES; WATER VAPOR
Broader Terms: ALLOYS; CHEMICAL REACTIONS; CORROSION; ENERGY SOURCES; FLUIDS; FUELS; GASES; MATERIALS; REACTOR MATERIALS; VAPORS

35/7,DE/8 (Item 3 from file: 103)
DIALOG(R)File 103:Energy SciTec

03434581 JPN-92-011448; EDB-93-013457
Title: Cladding tube for nuclear fuel and manufacturing method thereof
Author(s)/Editor(s): Oe, Akira.
Corporate Source: Nuclear Fuel Industries Ltd., Tokyo (Japan)
Patent No.: JP 4-128687 A
Patent Assignee(s): Nuclear Fuel Industries Ltd., Tokyo (Japan)

Priority No.: JP 2-248908
Patent Date Filed: 20 Sep 1990
Publication Date: 30 Apr 1992
(7 p)
Language: Japanese
Availability: Available from JAPIO. Also available from INPADOC.
Abstract: A fuel rod cladding tube of a PWR type reactor comprises Sn: 0.9 to 1.2 wt%, Fe: 0.24 to 0.30 wt%, Cr: 0.13 to 0.19 wt%, Nb: 0.05 to 0.15 wt%, Ni: 0.005 to 0.020 wt%, O: 1000 to 1500 ppm, C: 100 to 200 ppm, Si: 50 to 200 ppm and the balance of a zirconium alloy made of Zr and inevitable impurities. Upon fabricating the alloy into a tube, an fr value at the inner surface of the cladding tube is controlled to 0.65 to 0.75, while setting the fabrication degree in the final cold working step, for example, to 60 to 70%. Further, an annealing index is controlled to 2 x 10[sup -18] [<=] [Sigma] Ai[<=] 5 x 10[sup -17] in an annealing step. With such procedures, the corrosion resistance is improved by decreasing the amount of Sn and adding a trace amount of Ni, in addition, corrosion resistance is also improved by adding a trace amount of Nb. At the same time, hydrogen absorption is also suppressed. Further, corrosion resistance is ensured by adding a greater amount of Si and applying annealing even if the annealing index is low. (T.M.).
Major Descriptors: *FUEL ASSEMBLIES -- FUEL CANS; *FUEL ASSEMBLIES -- FUEL RODS; *PWR TYPE REACTORS -- FUEL ASSEMBLIES
Descriptors: ABSORPTION; ANNEALING; COLD WORKING; CORROSION RESISTANCE; HYDROGEN; ROLLING; TIME DEPENDENCE; TIN ADDITIONS; ZIRCONIUM ALLOYS
Broader Terms: ALLOYS; ELEMENTS; ENRICHED URANIUM REACTORS; FABRICATION; FUEL ELEMENTS; HEAT TREATMENTS; MATERIALS WORKING; NONMETALS; POWER REACTORS; REACTOR COMPONENTS; REACTORS; SORPTION; THERMAL REACTORS; TIN ALLOYS; WATER COOLED REACTORS; WATER MODERATED REACTORS

35/7,DE/9 (Item 4 from file: 103)
DIALOG(R)File 103:Energy SciTec

03285168 JPN-92-002426; EDB-92-047925
Title: Method for manufacturing composite cladding tube for nuclear fuel
Author(s)/Editor(s): Abe, Hideaki; Kobayashi, Toshimi.
Corporate Source: Sumitomo Metal Industries Ltd., Osaka (Japan)
Patent No.: JP 3-199995 A
Patent Assignee(s): Sumitomo Metal Industries Ltd., Osaka (Japan)
Priority No.: JP 1-341685
Patent Date Filed: 27 Dec 1989
Publication Date: 30 Aug 1991
(6 p)
Language: In Japanese
Availability: Available from JAPIO. Also available from INPADOC.
Abstract: In the present invention, a composite cladding tube for nuclear fuels having a double structure comprising a pure zirconium metal layer joined to the inner surface of a zirconium alloy tube is manufactured at high quality. That is, an intermediate annealing conducted during the repeating cold rolling fabrication for a zirconium alloy tube having a pure zirconium metal layer joined at the inner surface is conducted at a temperature of 450 to 580degC, which is kept for 1 to 4 hours, while the last annealing conducted after completion of the final cold rolling fabrication is conducted at temperature of 550 to 585degC, which is kept for 1 to 4 hours. Alternatively, annealing is applied to a zirconium alloy tube having a pure zirconium metal layer joined at the inner surface at 450 to 580degC for 1 to 4 hours and, thereafter,

it is subjected to cold rolling fabrication. According to the manufacturing method of the present invention, fine defects are not caused to the pure zirconium metal layer at the inner surface, or cause, if any, only shallow defects that can be eliminated. Accordingly, the production yield can be improved. (I.S.).
Major Descriptors: *FUEL CANS -- FABRICATION
Descriptors: ANNEALING; BWR TYPE REACTORS; COLD WORKING; MECHANICAL PROPERTIES; NUCLEAR FUELS; ROLLING; ZIRCONIUM; ZIRCONIUM BASE ALLOYS
Broader Terms: ALLOYS; ELEMENTS; ENERGY SOURCES; ENRICHED URANIUM REACTORS; FABRICATION; FUELS; HEAT TREATMENTS; MATERIALS; MATERIALS WORKING; METALS; POWER REACTORS; REACTOR MATERIALS; REACTORS; THERMAL REACTORS; TRANSITION ELEMENTS; WATER COOLED REACTORS; WATER MODERATED REACTORS; ZIRCONIUM ALLOYS

35/7,DE/10 (Item 5 from file: 103)
DIALOG(R)File 103:Energy SciTec

02011080 EDB-87-139033
Author(s): Inagaki, M.; Akabori, K.; Nakajima, J.
Title: Neuclear fuel cladding tube and its manufacture
Patent No.: JP 61-233391 A
Patent Assignee(s): Hitachi Ltd., Tokyo, Japan
Patent Date Filed: Filed date 9 Apr 1985
Publication Date: 17 Oct 1986
p 4
Note: JP patent application 60-74904
Language: Japanese
Abstract: Purpose: To overcome the problems of damage, particularly, stress corrosion crackings due to interactions between fuel cladding tubes and nuclear fuels thereby making the tubes endurable for a long period of use. Method: A liner layer comprising a zirconium-based alloy is metalluriqically joined to the surface of a tube made of a zirconium-based alloy. The liner layer is applied with a heat treatment of heating to a temperature range for (..cap alpha.. + ..beta..) phase, forming the ..beta.. phase at least to a portion in the ..cap alpha.. phase crystal grain boundary followed by quenching. In this way, the crystal grain boundary of the liner layer is formed as a super-saturated solid-solution phase to obtain a layer in which no intermetallic compound is present or the amount thereof is reduced in the crystal grain boundary. Such a fuel cladding tube is highly reliable against destruction, shows no risk for the tube damage even at an increased burnup degree, by which improvement can be made for increasing the reactor operation cycle and utilizing efficiency.
Major Descriptors: *FUEL CANS -- DESIGN; *FUEL CANS -- FABRICATION
Descriptors: ANNEALING; CHROMIUM ADDITIONS; FUEL-CLADDING INTERACTIONS; GRAIN BOUNDARIES; INTERMETALLIC COMPOUNDS; LINERS; ZIRCONIUM BASE ALLOYS
Broader Terms: ALLOYS; CHROMIUM ALLOYS; CRYSTAL STRUCTURE; HEAT TREATMENTS; MICROSTRUCTURE; ZIRCONIUM ALLOYS

35/7,DE/11 (Item 6 from file: 103)
DIALOG(R)File 103:Energy SciTec

01819494 AIX-17-059727; EDB-86-143368
Author(s): Takase, Iwao; Yoshida, Toshimi; Ikeda, Shinzo; Masaoka, Isao; Nakajima, Junjiro

Title: Nuclear fuel cladding tube and its fabrication
Corporate Source: Hitachi Ltd., Tokyo (Japan)
Patent No.: JP 60-165580 A
Patent Assignee(s): Hitachi Ltd., Tokyo, Japan
Patent Date Filed: Filed date 8 Feb 1984
Publication Date: 28 Aug 1985
p 10
Note: JP patent application 59-19980
Language: Japanese
Availability: JAPIO. Also available from INPADOC.
Abstract: The purpose of this patent is to manufacture fuel cladding tube made of zirconium-based alloy resistant to nodular corrosions water and steams at high temperature and less sensitive to stress corrosion cracks due to iodine or the like. In light water or heavy water reactor fuels, there are problems such as reduction tube wall thickness caused by the corrosion to the outside and stress corrosion cracks caused by the emission gas and pellet sintering to the inside of fuel cladding tubes. In order to increase the resistance to them, heating upon hardening to ..beta.. phase or (..beta.. + ..cap alpha..) phase region is carried out while cooling the inner surface of the tube by rapid quenching so as to avoid the hardening to the inner surface of the tube. After the hot plastic fabrication, annealing is carried out while making the temperature slope between the inside and the outside of the pipe such that the inside of the fuel cladding tube is at a temperature higher than the re-crystallization point of the alloy and the inside of the can is at a temperature lower than the re-crystallization point. In this method, the outer surface of the fuel cladding tube has a texture having hardened structure and the inner surface has a completely re-crystallized texture, thereby providing excellent nodular corrosion resistance and stress corrosion crack resistance.
Major Descriptors: *FUEL RODS -- CORROSION RESISTANCE; *FUEL RODS -- FABRICATION
Descriptors: ANNEALING; CRACKS; PITTING CORROSION; QUENCH HARDENING; RECRYSTALLIZATION; STEAM; STRESS CORROSION; WATER; ZIRCONIUM BASE ALLOYS
Broader Terms: ALLOYS; CHEMICAL REACTIONS; CORROSION; FUEL ELEMENTS; HARDENING; HEAT TREATMENTS; HYDROGEN COMPOUNDS; OXYGEN COMPOUNDS; REACTOR COMPONENTS; ZIRCONIUM ALLOYS

35/7,DE/12 (Item 7 from file: 103)
DIALOG(R)File 103:Energy SciTec

01409701 AIX-15-028538; EDB-84-107501
Title: New production facilities for nuclear fuel cladding tubes
Author(s): Konishi, Takao; Inoue, Mamoru; Matsuda, Katsuhiko; Kojima, Tatsuhisa (Sumitomo Metal Industries Ltd., Amagasaki, Hyogo (Japan). Steel Tube Works)
Source: Sumitomo Kinzoku (Japan) v 34:1. Coden: SUKIA
Publication Date: Jan 1982
p 161-167
Language: Japanese
Abstract: In order that we may meet the growing demand for nuclear fuel cladding tubes for nuclear power generation, we established the new Zircaloy and stainless steel fuel cladding tube plant on the premises of our Steel Tube Works. The new plant named ''Precision Tube Making Plant'' is equipped with a complete series of highly advanced facilities for tube making, finishing and inspecting, and its operation environment is especially kept clean to make severe quality control. In

September 1980, the new plant started operating successfully. Although its present production capacity is 300,000 - 400,000 m/year, we can expand the capacity to the scale of 1,000,000 m/year. This paper gives outlines of manufacturing process and of main manufacturing and inspection equipment for nuclear fuel cladding tubes.;
Major Descriptors: *FUEL CANS -- INDUSTRIAL PLANTS; *FUEL CANS -- MANUFACTURING
Descriptors: CLADDING; COLD WORKING; EQUIPMENT; FLOWSHEETS; FUEL FABRICATION PLANTS; INSPECTION; JAPAN; QUALITY CONTROL; STAINLESS STEELS; TUBES; ZIRCALOY
Broader Terms: ALLOYS; ASIA; CHROMIUM ALLOYS; CONTROL; CORROSION RESISTANT ALLOYS; DEPOSITION; DIAGRAMS; FABRICATION; IRON ALLOYS; IRON BASE ALLOYS; MATERIALS WORKING; NUCLEAR FACILITIES; STEELS; SURFACE COATING; TIN ALLOYS; ZIRCONIUM ALLOYS; ZIRCONIUM BASE ALLOYS

35/7,DE/13 (Item 8 from file: 103)
DIALOG(R)File 103:Energy SciTec

00638258 AIX-11-514265; EDB-80-077783
Author(s): Shirokane, M.; Chida, T.; Shirai, H.; Watanabe, K.
Title: Method of fabricating fuel rod end plug (Patent)
Patent No.: JP 54-47085 A
Patent Assignee(s): Toshiba Corp., Kawasaki, Kanagawa (Japan)
Publication Date: 13 Apr 1979
p 3
Language: Japanese
Abstract: Purpose: To increase the reliability on the weldability of nuclear fuel pellets with the cladding tube thereby to improve the end plug characteristic. Method: A columnar material made of a zirconium alloy is inserted into a die, and applied with pressure by a punch from the upper part of the columnar material in the heating state, the end part of said columnar material being subjected to extraction forging in the ..cap alpha.. region. By this operation, an end plug whose small diameter portion to be fitted in the fuel rod supporting tool has been shaped is fabricated. Thus, there is produced no residual strain due to the processing at the head of the end plug which is the largest caliber portion, and hence the desired purpose can be achieved.;
Major Descriptors: *FUEL PELLETS -- WELDABILITY; *ZIRCALOY 2 -- WELDABILITY
Descriptors: CLOSURES; FABRICATION; FORGING; FUEL CANS; FUEL ELEMENTS
Broader Terms: ALLOYS; CHROMIUM ADDITIONS; CHROMIUM ALLOYS; FABRICATION; IRON ADDITIONS; IRON ALLOYS; MATERIALS WORKING; PELLETS; REACTOR COMPONENTS; TIN ALLOYS; ZIRCALOY; ZIRCONIUM ALLOYS; ZIRCONIUM BASE ALLOYS

35/7,DE/14 (Item 1 from file: 347)
DIALOG(R)File 347:JAPIO

06082244
MANUFACTURE OF PIPE FOR CLADDING NUCLEAR FUEL

PUB. NO.: 11-023758 [JP 11023758 A]
PUBLISHED: January 29, 1999 (19990129)
INVENTOR(s): ABE HIDEAKI
TAKEDA KIYOKO
APPLICANT(s): SUMITOMO METAL IND LTD
APPL. NO.: 09-182291 [JP 97182291]

FILED: July 08, 1997 (19970708)

ABSTRACT

PROBLEM TO BE SOLVED: To reduce manufacturing costs by performing intermediate cold rolling working once to a material pipe having a recrystallization structure under specific conditions and performing softening heat treatment, final cold rolling, and annealing heat treatment to the formed object under specific conditions.

SOLUTION: A pipe for cladding nuclear fuel consisting of zirconium based alloy being used for a boiling water type nuclear reactor is manufactured of a solid material pipe and a double material pipe in hot extrusion. For example, the material pipes that have, for example, an outer diameter of approximately 63.5 mm and a thickness of approximately 10.9 mm are subjected to recrystallization annealing heat treatment under specific conditions and are subjected to intermediate cold rolling machining with a section reduction rate to be 90% or higher. Then, softening annealing at 540-680°C, final cold rolling machining with a section reduction rate of 60-85%, and final annealing heat treatment at 550-600°C are performed, thus obtaining a pipe for cladding nuclear fuel with improve mechanical property, where the outer diameter is approximately 11-13 mm and the thickness is approximately 0.7-0.9 mm. The intermediate cold rolling machine and softening annealing with large treatment costs are performed only once, thus reducing manufacturing costs.

35/7,DE/15 (Item 2 from file: 347)
DIALOG(R)File 347:JAPIO

05990646
ZIRCONIUM ALLOY EXCELLENT IN COLD WORKABILITY AND CORROSION RESISTANCE, DUPLEX TUBE FOR CLADDING NUCLEAR FUEL USING THIS ALLOY AND PRODUCTION THEREOF

PUB. NO.: 10-273746 [JP 10273746 A]
PUBLISHED: October 13, 1998 (19981013)
INVENTOR(s): ABE HIDEAKI
APPLICANT(s): SUMITOMO METAL IND LTD [000211] (A Japanese Company or Corporation), JP (Japan)
APPL. NO.: 09-013658 [JP 9713658]
FILED: January 28, 1997 (19970128)
JAPIO CLASS: 12.3 (METALS -- Alloys); 12.2 (METALS -- Metallurgy & Heat Treating); 23.1 (ATOMIC POWER -- General)

ABSTRACT

PROBLEM TO BE SOLVED: To obtain a zirconium alloy excellent in cold-workability and corrosion resistance and suitable as the material for nuclear fuel cladding duplex tube of a water-cooled type nuclear reactor by specifying its composition composed of Sn, Fe, Cr and Zr and further incorporated with Ni and Nb according to necessary.

SOLUTION: This is a low Sn-Zr alloy having a composition containing, by weight, 0.30 to 0.70% Sn, 0.20 to 0.25% Fe and 0.10 to 0.15% Cr, furthermore containing, at need, one or both of 0.005 to 0.05% Ni and 0.05 to 0.20% Nb, and the balance Zr with inevitable impurities, and in which cracks and strains are not generated even by cold rolling in which the

reduction of cross-sectional area is regulated to about >=80%. By forming an outer tube by this Zr alloy and making an inner tube of a high Sn-Zr alloy containing 1.2 to 1.7% Sn, the duplex tube for cladding nuclear fuel excellent in corrosion resistance in the outer face in which CSR value defined by the formula of CSR=.epsilon.(sub s)/.epsilon.(sub r) (.epsilon.(sub s) and .epsilon.(sub r) denote the strains in the circumferential direction and to strains in the thickness direction) is equal to that of a solid tube of the Zr alloy same as that of the inner tube and having high strength in the whole body can securely be obtained at a low cost.

35/7,DE/16 (Item 3 from file: 347)
DIALOG(R)File 347:JAPIO

05790590
HIGHLY ANTICORROSIVE CLADDING TUBE FOR NUCLEAR FUEL, SPACER, CHANNEL BOX, FUEL ASSEMBLY THEREOF AND METHOD FOR MANUFACTURING IT

PUB. NO.: 10-073690 [JP 10073690 A]
PUBLISHED: March 17, 1998 (19980317)
INVENTOR(s): INAGAKI MASATOSHI
TAKASE IWAO
SUGANO MASAYOSHI
KUNIYA JIRO
AKAHORI KIMIHIKO
MASAOKA ISAO
MAKI HIDEO
NAKAJIMA JUNJIRO
APPLICANT(s): HITACHI LTD [000510] (A Japanese Company or Corporation), JP (Japan)
APPL. NO.: 09-175505 [JP 97175505]
FILED: July 01, 1997 (19970701)
JAPIO CLASS: 23.1 (ATOMIC POWER -- General); 12.2 (METALS -- Metallurgy & Heat Treating); 12.3 (METALS -- Alloys)
JAPIO KEYWORD:R002 (LASERS); R003 (ELECTRON BEAM)

ABSTRACT

PROBLEM TO BE SOLVED: To improve crossion resistance and hydrogen absorption characteristics by composing at least one of a cladding tube for nuclear fuel, a spacer for a nuclear fuel assembly and a channel box for a nuclear fuel assembly of an alloy made of specific materials in specific proportions.

SOLUTION: In a fuel assembly for a nuclear reactor, at least one of a cladding tube, a spacer and a channel box is made of an alloy of the following composition in weight percentage. The alloy contains tin of about 1.20 to 2%, iron of about 0.20 to 0.5%, chromium of about 0.05 to 0.15% and nickel of about 0.03% to 0.16%, with the rest constituted materially of zirconium. These metal elements constitute a zirconium-group alloy where the proportion of iron to nickel is about 1.4 to 10. After hot plastic working, this alloy is heated and maintained for a short time in a .beta. phase or a temperature range containing .alpha. and .beta. phases and then is rapidly cooled before cold working. Subsequently, cool plastic working and anneal working are repeated alternately and shaping work is done by forming a thin-wall material made of a zirconium-group alloy. This makes it possible to obtain an alloy which is excellent in corrosion resistance and absorbs little hydrogen and enables the higher burn-up of fuel.

35/7,DE/17 (Item 4 from file: 347)
DIALOG(R)File 347:JAPIO

04816954
PRODUCTION OF NUCLEAR FUEL CLADDING PIPE

PUB. NO.: 07-109554 [JP 7109554 A]
PUBLISHED: April 25, 1995 (19950425)
INVENTOR(s): HISATSUNE YOSHIMI
HIGASHINAKAGAHA EMIKO
ARAI SHINJI
IKEDA TADAHIRO
APPLICANT(s): TOSHIBA CORP [000307] (A Japanese Company or Corporation), JP (Japan)
APPL. NO.: 05-276000 [JP 93276000]
FILED: October 08, 1993 (19931008)
JAPIO CLASS: 12.2 (METALS -- Metallurgy & Heat Treating); 23.1 (ATOMIC POWER -- General)

ABSTRACT

PURPOSE: To produce a nuclear fuel cladding pipe excellent in uniform corrosion resistance in an external pipe and excellent in corrosion resistance in a steam environment even in a linear layer by providing a special heat treating operation in the producing process.

CONSTITUTION: This method comprises stage in which a Zr liner layer is formed on the inside face of an external pipe constituted of a Zr alloy and a stage in which the external pipe and Zr liner are rapidly heated to a high temperature in an .alpha. region, is held and is thereafter rapidly cooled. Thus, the nuclear fuel cladding pipe in which uniform corrosion resistance is improved without deteriorating its nodular corrosion resistance in the external pipe and corrosion resistance in a steam environment is excellent even in the liner layer can be produced.

35/7,DE/18 (Item 5 from file: 347)
DIALOG(R)File 347:JAPIO

03373348
CORROSION RESISTING ZIRCONIUM-BASE ALLOY AND ITS PRODUCTION

PUB. NO.: 03-036248 [JP 3036248 A]
PUBLISHED: February 15, 1991 (19910215)
INVENTOR(s): EITO YOSHINORI
APPLICANT(s): NIPPON NUCLEAR FUEL DEV CO LTD [472479] (A Japanese Company or Corporation), JP (Japan)
APPL. NO.: 01-171139 [JP 89171139]
FILED: July 04, 1989 (19890704)
JAPIO CLASS: 12.2 (METALS -- Metallurgy & Heat Treating); 12.6 (METALS -- Surface Treatment); 23.1 (ATOMIC POWER -- General)

ABSTRACT

PURPOSE: To provide nodular corrosion resistance suitable for a core internal structure material by irradiating a film layer consisting of the prescribed elements formed on the surface of the alloy for nuclear reactor with a corpuscular beam and allowing a region deficient in the solid solution components of alloying elements to disappear.

CONSTITUTION: An ingot is prepared by adding the prescribed alloying elements (tin, iron, Cr, Ni, etc.) to pure-Zr sponge for nuclear reactor fuel clad pipe or core internal structure material and carrying out arc melting. The above ingot is subjected to .beta.- forging, solution treatment, and .alpha.-forging and is formed into a tube stock by means of hot working, which is formed into the desired thin-wall finished product while exerting a repetition of cold rolling and annealing. In the above method, after the final annealing stage, a film layer consisting of the prescribed elements (Co, Ni, iron, etc.) is formed by a plating method, to which corpuscular beam irradiation is applied. By this method, the region deficient in the solid solution components of alloying elements in the vicinity of the surface of the Zr-base alloy can be allowed to disappear, and as a result, nodular corrosion resulting from the presence of the above deficient region can be inhibited.

35/7,DE/19 (Item 6 from file: 347)
DIALOG(R)File 347:JAPIO

02039251
SUPERPLASTIC ZIRCONIUM ALLOY AND ITS MANUFACTURE

PUB. NO.:	61-253351 [JP 61253351 A]
PUBLISHED:	November 11, 1986 (19861111)
INVENTOR(s):	KUBO TOSHIO MOTOMIYA TAKEO
APPLICANT(s):	NIPPON NUCLEAR FUEL DEV CO LTD [472479] (A Japanese Company or Corporation), JP (Japan)
APPL. NO.:	60-092821 [JP 8592821]
FILED:	April 30, 1985 (19850430)
JAPIO CLASS:	12.2 (METALS -- Metallurgy & Heat Treating); 12.3 (METALS -- Alloys); 23.1 (ATOMIC POWER -- General)

ABSTRACT

PURPOSE: To manufacture a superplastic Zr alloy having superior resistance to stress corrosion cracking and corrosion by hardening and aging a Zr alloy for the constituent parts of a nuclear fuel assembly for a nuclear fission reactor under specified conditions.

CONSTITUTION: When a pipe 1 for cladding fuel pellets 2 for a nuclear fission reactor is made of a Zr alloy, the pipe 1 causes corrosion by a coolant and stress corrosion cracking by the thermal expansion of the pellets 2. In order to prevent the corrosion and stress corrosion cracking and to provide superplasticity, the Zr alloy is subjected to solution heat treatment in the .alpha.+.beta. phase region (800-950 deg.C) of the alloy, cooled to 700 deg.C at <=100 deg.C/sec cooling rate, and aged at a proper temperature of 550-700 deg.C in the .alpha.-phase region. By this heat treatment, a cladding pipe of a Zr alloy causing no stress corrosion cracking and having superior resistance to corrosion by a coolant is obtained

35/7,DE/20 (Item 7 from file: 347)
DIALOG(R)File 347:JAPIO

01965760
MANUFACTURE OF NUCLEAR FUEL CLADDING TUBE MADE OF ZR BASE ALLOY

PUB. NO.: 61-179860 [JP 61179860 A]
PUBLISHED: August 12, 1986 (19860812)
INVENTOR(s): ABE HIDEAKI
HONCHI MASAHIRO
APPLICANT(s): SUMITOMO METAL IND LTD [000211] (A Japanese Company or Corporation), JP (Japan)
APPL. NO.: 60-001177 [JP 851177]
FILED: January 08, 1985 (19850108)
JAPIO CLASS: 12.2 (METALS -- Metallurgy & Heat Treating); 23.1 (ATOMIC POWER -- General)

ABSTRACT
PURPOSE: To manufacture stably and surely a pipe superior in anisotropy and mechanical property of material characteristic, by performing finishing cold roll working and annealing under specified conditions, at manufacturing nuclear fuel cladding tube composed of Zr base alloy having a specified component composition
CONSTITUTION: In manufacturing the titled tube composed of, by weight ratio 1.20-1.70% Sn, 0.18-0.24% Fe, 0.07-0.13% Cr, under (Fe(%)+Cr(%))=0.28-0.37, and the balance Zr with inevitable impurities, finishing cold roll working is carried out under 1.5-3.5 working parameter (QE value), (3.3+0.16.ln QE)/4.8X100-90% working degree Rd. QE value is exhibited by a formula, Dm is average tube diameter before rolling, (dm) is said diameter after rolling, T is tube wall thickness before rolling, (t) is said thickness after rolling. Successive annealing is performed as stress removal annealing at 430-500 deg.C.

35/7,DE/21 (Item 8 from file: 347)
DIALOG(R)File 347:JAPIO

01965759
MANUFACTURE OF NUCLEAR FUEL CLADDING TUBE MADE OF ZR BASE ALLOY

PUB. NO.: 61-179859 [JP 61179859 A]
PUBLISHED: August 12, 1986 (19860812)
INVENTOR(s): ABE HIDEAKI
HONCHI MASAHIRO
APPLICANT(s): SUMITOMO METAL IND LTD [000211] (A Japanese Company or Corporation), JP (Japan)
APPL. NO.: 60-001176 [JP 851176]
FILED: January 08, 1985 (19850108)
JAPIO CLASS: 12.2 (METALS -- Metallurgy & Heat Treating); 23.1 (ATOMIC POWER -- General)

ABSTRACT
PURPOSE: To manufacture stably, surely a tube superior in anisotropy and mechanical property of material characteristic, by performing cold roll working and annealing under specified conditions, at manufacturing nuclear fuel cladding tube composed of Zr base alloy having a specified component composition
CONSTITUTION: In manufacturing the titled tube composed of, by weight ratio 1.20-1.70% Sn, 0.07-0.20% Fe, 0.05-0.15% Cr, 0.03-0.08 Ni under (Fe(%)+Cr(%)+Ni(%))=0.18-0.38, and the balance Zr with inevitable impurities, finishing clf roll working is carried out under 1.5-3.5 working parameter (QE value), <=(6.6+0.50.ln QE)/8.4X100 working degree (Rd)(%). QE value is exhibited by a formula, Dm is average tube diameter before rolling, dm is said diameter after rolling, T is tube wall thickness before rolling, (t) is said thickness after rolling. Successive annealing is

performed as recrystallization annealing at >=550 deg.C.

35/7,DE/22 (Item 9 from file: 347)
DIALOG(R)File 347:JAPIO

01795464
MANUFACTURE OF COMPOSITE CLAD PIPE FOR NUCLEAR REACTOR

PUB. NO.: 61-009564 [JP 61009564 A]
PUBLISHED: January 17, 1986 (19860117)
INVENTOR(s): ASAHI KAZUMI
APPLICANT(s): NIPPON NUCLEAR FUEL DEV CO LTD [472479] (A Japanese Company or Corporation), JP (Japan)
APPL. NO.: 59-129361 [JP 84129361]
FILED: June 25, 1984 (19840625)
JAPIO CLASS: 12.2 (METALS -- Metallurgy & Heat Treating); 12.5 (METALS -- Working); 23.1 (ATOMIC POWER -- General)

ABSTRACT
PURPOSE: To improve the mechanical strength and corrosion resistance by forming a layer of a uniform thickness having a .beta.-hardened structure on the outside of a Zr alloy layer having an .alpha.-tempered structure.

CONSTITUTION: An .alpha.-tempered pierced Zr alloy billet 1 having an .alpha.-tempered structure and a .beta.-hardened pierced billet 2 having a .beta.-hardened structure are formed. The billet 1 is put in the billet 2 and united to one body by hot rolling mill 3 to manufacture a rough pipe 4 for a composite clad pipe. The pipe 4 is cold rolled to prescribed final dimensions to obtain a composite clad pipe for a nuclear reactor consisting of a Zr alloy layer 5 having an .alpha.-tempered structure and a layer 6 of a uniform thickness having a .beta.-hardened structure formed on the outside of the layer 5.

35/7,DE/23 (Item 10 from file: 347)
DIALOG(R)File 347:JAPIO

01743060
MANUFACTURE OF ZIRCONIUM ALLOY

PUB. NO.: 60-221560 [JP 60221560 A]
PUBLISHED: November 06, 1985 (19851106)
INVENTOR(s): NAKAJIMA JUNJIRO
SHINPO KATSUTOSHI
YASUDA TETSUO
ASANO RINICHI
INAGAKI MASATOSHI
APPLICANT(s): HITACHI LTD [000510] (A Japanese Company or Corporation), JP (Japan)
APPL. NO.: 59-077399 [JP 8477399]
FILED: April 16, 1984 (19840416)
JAPIO CLASS: 12.2 (METALS -- Metallurgy & Heat Treating); 23.1 (ATOMIC POWER -- General)

ABSTRACT
PURPOSE: To manufacture a nuclear fuel cladding pipe with high corrosion resistance by quenching a hot worked Zr alloy plate from a specified

temperature and subjecting it to cold plastic working and annealing plural times.

CONSTITUTION: When a fuel cladding pipe for a nuclear power plant is manufactured with a Zr alloy such as zircaloy-2, a Zr alloy ingot is worked into a plate by hot rolling or other process. This plate is hardened by heating to >=870c in a temperature range in which .alpha.- and .beta.-phase are included, holding at the temperature for >=30sec, and quenching to 500c at >=100c/sec cooling rate to form a fine acicular structue consisting of fine grains. The hardened plate is subjected to cold plastic working and annealing plural times to manufacture a nuclear fuel cladding pipe. This Zr alloy cladding pipe prevents nodular corrosion in a nuclear reactor for a long period.

35/7,DE/24 (Item 11 from file: 347)
DIALOG(R)File 347:JAPIO

01571655
PRODUCTION OF NUCLEAR FUEL CLADDING PIPE

PUB. NO.: 60-050155 [JP 60050155 A]
PUBLISHED: March 19, 1985 (19850319)
INVENTOR(s): HIGASHINAKAGAHA EMIKO
SATO KANEMITSU
KAWASHIMA JUNKO
KAMEI TOSHIO
APPLICANT(s): TOSHIBA CORP [000307] (A Japanese Company or Corporation), JP (Japan)
NIPPON ATOM IND GROUP CO LTD [352264] (A Japanese Company or Corporation), JP (Japan)
APPL. NO.: 58-157610 [JP 83157610]
FILED: August 29, 1983 (19830829)
JAPIO CLASS: 12.2 (METALS -- Metallurgy & Heat Treating); 23.1 (ATOMIC POWER -- General)

ABSTRACT
PURPOSE: To improve the corrosion resistance and mechanical characteristic of a Zr alloy pipe as a nuclear fuel cladding pipe by subjecting the Zr alloy pipe to a drawing stage until said pipe is reduced to the prescribed bore and wall thickness then heating quickly the pipe to a high temperature in an .alpha. region as a final heat treatment and cooling quickly the heated pipe immediately after holding for a short time.

CONSTITUTION: A hollow billet consisting of a Zr alloy is hot-extruded and is then subjected to a drawing stage consisting of cold working, by which the billet is reduced to the finishing bore and wall thickness. The drawing stage is accomplished in combination with annealing at the intermediate and the final annealing is accomplished with 3-4 passes. The Zr alloy pipe subjected to the final annealing is quickly heated at the surface part up to 780-860c in the .alpha. region by, for example, high-frequency heating and after the pipe is held for several seconds, the pipe is quickly cooled so that the residual stress remains on the surface part. The nuclear fuel cladding pipe having excellent resistance to nodular corrosion, an excellent mechanical characteristic and less bending and twisting is thus obtained

35/7,DE/25 (Item 12 from file: 347)

DIALOG(R)File 347:JAPIO

00395085
MANUFACTURING METHOD OF END PLUG OF NUCLEAR FUEL ROD

PUB. NO.: 54-047085 [JP 54047085 A]
PUBLISHED: April 13, 1979 (19790413)
INVENTOR(s): SHIROKANE MAKOTO
SENDA CHUICHI
SHIRAI HIDEO
WATANABE KUNIMICHI
APPLICANT(s): TOSHIBA CORP [000307] (A Japanese Company or Corporation), JP (Japan)
APPL. NO.: 52-112531 [JP 77112531]
FILED: September 21, 1977 (19770921)
JAPIO CLASS: 23.1 (ATOMIC POWER -- General); 12.5 (METALS -- Working)

ABSTRACT
PURPOSE: To heighten the reliablity of welding with a cladding tube of nuclear fuel pellets, and to improve the characteristic of an end plug, without causing a residual strain due to the processing of the head part of the end plug to be generated, by means of moulding the end part of a circular cylindrical material of Zr-series alloy by an extrusion precision casting in .alpha. zone.

CONSTITUTION: A circular cylindrical material 10 of Zr-series alloy is inserted in a ponch 11 and a dies 12, and pressure is exerted in a heated state from the upper part of the circular cylindrical material 10, thereby an end plug 4 is manufactured by means of forming the end part of the circular cylindrical material 10 into the small diameter part 5 which is engaged with a supporting tool of a nuclear fuel by an extrusion precision casting in .alpha. zone. A residual strain due to the processing of the head part 6 which is the max. aperture part of the end plug 4, does not been generated, thereby, the reliablity of welding an fxing with the cladding tube into which the fuel pellets is inserted, is heightened, and the characeristic of the end plug can be improved

35/7,DE/26 (Item 1 from file: 351)
DIALOG(R)File 351:Derwent WPI

011247439
WPI Acc No: 1997-225342/199720
Zirconium@ alloy tubing fabrication used for BWR and PWR - by coarsening annealing alloy billet, forming into tube, heat treating outside of tube while cooling inside, then rapidly quenching outside of tube giving crack and corrosion resistance
Patent Assignee: GENERAL ELECTRIC CO (GENE)
Inventor: ADAMSON R B; POTTS G A
Number of Countries: 001 Number of Patents: 001
Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week	
US 5618356	A	19970408	US 9352791	A	19930423	199720	B
			US 9352793	A	19930423		
			US 95489597	A	19950612		

Priority Applications (No Type Date): US 95489597 A 19950612; US 9352791 A 19930423; US 9352793 A 19930423

Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
US 5618356	A		13	C22F-001/18	CIP of application US 9352791
					CIP of application US 9352793
					CIP of patent US 5437747
					CIP of patent US 5519748

Abstract (Basic): US 5618356 A

Preparing a corrosion resistant Zr alloy tube comprises coarsening annealing at 621 deg. C, converting alloy billet to alloy tube, heat treating an outer region of the tube at alpha + beta temp. region while cooling inner region of the tube (104), rapidly quenching outer region and performing one or more cold working steps, each cold work step being followed by an anneal step at a temp of 576 deg. C.

Further claimed is a process similar to the above where the cold work steps are dispersed throughout the process. Also claimed is a process for forming a barrier containing tube similar to the above.

USE - The zircalloy cladding is used in the nuclear fuel industry, particularly for BWR and PWR

ADVANTAGE - Tube is resistant to axial crack propagation, crack initiation and nodular corrosion.

Dwg.3/5

Title Terms: ZIRCONIUM; ALLOY; TUBE; FABRICATE; BWR; PWR; COARSE; ANNEAL; ALLOY; BILLET; FORMING; TUBE; HEAT; TREAT; TUBE; COOLING; RAPID; QUENCH; TUBE; CRACK; CORROSION; RESISTANCE

Derwent Class: K05; M29; X14

International Patent Class (Main): C22F-001/18

35/7,DE/27 (Item 2 from file: 351)

DIALOG(R)File 351:Derwent WPI

010750189

WPI Acc No: 1996-247144/199625

Corrosion-resistant zirconium@ alloy mfr. - by beta-treatment, hot working, repeatedly cold working and process annealing, e.g. used for LWR nuclear fuel cladding tubes

Patent Assignee: SUMITOMO METAL IND LTD (SUMQ)

Number of Countries: 001 Number of Patents: 001

Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week	
JP 8100231	A	19960416	JP 94236512	A	19940930	199625	B

Priority Applications (No Type Date): JP 94236512 A 19940930

Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
JP 8100231	A		5	C22C-016/00	

Abstract (Basic): JP 8100231 A

The Zr-alloy is made by hot working after beta-treatment, repeatedly cold working and process annealing, and final annealing after final cold working. The process annealing before final cold working is performed at temps. of at least 750deg.C.

USE - Used for fuel cladding tubes of light water reactors.

ADVANTAGE - Good resistance to general corrosion.

Dwg.0/1

Title Terms: CORROSION; RESISTANCE; ZIRCONIUM; ALLOY; MANUFACTURE; BETA; TREAT; HOT; WORK; REPEAT; COLD; WORK; PROCESS; ANNEAL; NUCLEAR; FUEL; CLAD; TUBE

Index Terms/Additional Words: LIGHT; WATER; REACTOR
Derwent Class: K05; M26; X14
International Patent Class (Main): C22C-016/00
International Patent Class (Additional): C22F-001/18

35/7,DE/28 (Item 3 from file: 351)
DIALOG(R)File 351:Derwent WPI

010699819
WPI Acc No: 1996-196774/199620
High corrosion resistant zirconium alloy prodn. - by soln. treating zirconium alloy, hot and/or cold working, and heat treatment at specific range, used to clad tubes of nuclear power plants
Patent Assignee: SUMITOMO METAL IND LTD (SUMQ)
Number of Countries: 001 Number of Patents: 001
Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week	
JP 8067954	A	19960312	JP 94200871	A	19940825	199620	B

Priority Applications (No Type Date): JP 94200871 A 19940825
Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
JP 8067954	A		6	C22F-001/18	

Abstract (Basic): JP 8067954 A
The Zr-alloy is made by soln. treating a Zr-alloy stock contg. (by wt.) 0.4-1.7% Sn, 0.25-0.75% Fe, 0.05-0.30% Cr, 0-0.10% Ni, and 0-1.0% Ni, followed by hot working and/or cold working, in which heat treatment at alpha-phase range accompanied with the working is performed in a range where the total value of heat treatment parameter (Ai) in each heat treatment is 8.5 x 10-16-2.1 x 10-14, (where Ai = ti x exp(-650/RTi), ti = heat treating time in i-th heat treating process, and Ti = heat treating temps. in i-th heat treating process, and R = gas constant (cal/mol.K)).
USE - For fuel cladding tubes of nuclear power plants.
Dwg.0/0
Title Terms: HIGH; CORROSION; RESISTANCE; ZIRCONIUM; ALLOY; PRODUCE; SOLUTION; TREAT; ZIRCONIUM; ALLOY; HOT; COLD; WORK; HEAT; TREAT; SPECIFIC ; RANGE; CLAD; TUBE; NUCLEAR; POWER; PLANT
Derwent Class: K05; M26; M29
International Patent Class (Main): C22F-001/18
International Patent Class (Additional): C22C-016/00

35/7,DE/29 (Item 4 from file: 351)
DIALOG(R)File 351:Derwent WPI

010436608
WPI Acc No: 1995-337924/199544
Nuclear fuel cladding tube mfr. - involves annealing to diffuse alloying elements from lining into zirconium@ barrier layer
Patent Assignee: GENERAL ELECTRIC CO (GENE)
Inventor: ADAMSON R B; ARMIJO J S; LUTZ D R
Number of Countries: 004 Number of Patents: 006
Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week	
DE 19509049	A1	19950928	DE 1009049	A	19950314	199544	B

SE 9500969	A	19950922	SE 95969	A	19950320	199549
US 5469481	A	19951121	US 9391672	A	19930714	199601
			US 94215456	A	19940321	
JP 8043567	A	19960216	JP 9560179	A	19950320	199617
JP 2815551	B2	19981027	JP 9560179	A	19950320	199848
SE 510099	C2	19990419	SE 95969	A	19950320	199922

Priority Applications (No Type Date): US 94215456 A 19940321; US 9391672 A 19930714

Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
DE 19509049	A1		9	G21C-021/02	
US 5469481	A		9	G21C-003/00	CIP of application US 9391672 CIP of patent US 5383228
JP 8043567	A		8	G21C-003/06	
JP 2815551	B2		8	G21C-003/06	Previous Publ. patent JP 8043567
SE 9500969	A			G21C-003/20	
SE 510099	C2			G21C-003/20	

Abstract (Basic): DE 19509049 A

Prodn. of a cladding tube, with an outer substrate, an intermediate Zr barrier layer and an inner alloyed Zr lining, involves (a) joining the outer surface of the barrier layer to the inner surface of the substrate; (b) joining the outer surface of the inner lining to the inner surface of the barrier layer; and (c) diffusion annealing to diffuse alloying elements from the lining into the barrier layer for forming a diffusion layer having an alloying element concn. which decreases from the inner surface of the barrier layer to a zero concn. location within the barrier layer. Also claimed is a similar process in which steps (a)-(c) are followed by (d) cold working with intermediate stress-relief or recrystallisation anneals; and (e) heating no more than the outer 33% of the outer substrate in the alpha & beta-phases or the beta-phase region and cooling to produce a distribution of fine pptes. in the outer region of the substrate.

USE - As a nuclear reactor fuel cladding tube.

ADVANTAGE - The barrier layer is alloyed at its inner surface (facing the fuel), to provide corrosion resistance, and is unalloyed at its outer region to retain sufficient ductility to protect against damage caused by fuel pellet-cladding interaction.

Dwg.2/2

Abstract (Equivalent): US 5469481 A

A method of making a cladding tube having an outer substrate, an intermediate zirconium barrier layer, and a zirconium-based inner liner having alloying elements, the substrate, barrier layer, and inner liner each having interior and exterior circumferential surfaces, the method comprising the following steps:

(a) bonding the zirconium barrier layer exterior circumferential surface to the substrate interior circumferential surface;

(b) bonding the inner liner outer circumferential surface to the zirconium barrier layer inner circumferential surface; and

(c) conducting a diffusion anneal after steps (a) and (b) at a time and temperature sufficient to cause the alloying elements from the inner liner to diffuse into the barrier layer to form a diffusion layer containing a concentration of alloying elements that decreases from the interior circumferential surface of the barrier layer to a location interior to the barrier layer where there is substantially no alloying elements, wherein the alloying elements in the diffusion layer impart corrosion resistance to the barrier layer.

Dwg.0/2

Title Terms: NUCLEAR; FUEL; CLAD; TUBE; MANUFACTURE; ANNEAL; DIFFUSION;

ALLOY; ELEMENT; LINING; ZIRCONIUM; BARRIER; LAYER
Derwent Class: K05; M26; P71; X14
International Patent Class (Main): G21C-003/00; G21C-003/06; G21C-003/20; G21C-021/02
International Patent Class (Additional): B30B-012/00; G21D-001/00

35/7,DE/30 (Item 5 from file: 351)
DIALOG(R)File 351:Derwent WPI

010066940
WPI Acc No: 1994-334653/199442
Fabrication of zirconium@ alloy cladding tube - produces a component which has a high resistance to crack propagation.
Patent Assignee: GENERAL ELECTRIC CO (GENE)
Inventor: ADAMSON R B; POTTS G A
Number of Countries: 010 Number of Patents: 006
Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week	
EP 622470	A1	19941102	EP 94302539	A	19940411	199442	B
TW 233361	A	19941101	TW 94100032	A	19940104	199503	
JP 7090522	A	19950404	JP 9481201	A	19940420	199522	
US 5437747	A	19950801	US 9352791	A	19930423	199536	
US 5681404	A	19971028	US 9352791	A	19930423	199749	
			US 95385807	A	19950209		
MX 186392	B	19971013	MX 942968	A	19940422	199901	

Priority Applications (No Type Date): US 9352791 A 19930423; US 95385807 A 19950209
Cited Patents: EP 425465; US 4576654; US 4671826; US 4718949
Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
EP 622470	A1	E	14	C22F-001/18	
Designated States (Regional): CH DE ES IT LI SE					
TW 233361	A			G21C-003/00	
JP 7090522	A		11	C22F-001/18	
US 5437747	A		13	C22F-001/18	
US 5681404	A		11	C22F-001/18	Div ex application US 9352791
					Div ex patent US 5437747
MX 186392	B			C22F-001/018	

Abstract (Basic): EP 622470 A

A method of processing a zirconium alloy tube is disclosed. A coarsening anneal is performed at 621 deg. C. min. for 1-100 hours and an outer region of the tube is selectively heat treated to at least the alpha plus beta region whilst cooling the inner region of the tube and rapidly quenching the outer region. One or more cold work operations are carried out each followed by an annealing heat treatment at 576 deg. C. Min.

The zirconium alloy is pref. selected from Zircalloy-2, Zircalloy-4 and Zirlo. The selective heat-treatment step heats the outer region of the tube to 927 deg. C. approx. The accumulated normalised annealing time is approx. 5 multiply 10-17 hours. The outer region may be selectively heated by an induction coil heater.

USE -For fabricating zirconium alloy cladding for use in nuclear fuel elements.

ADVANTAGE - The cladding is resistant to axial crack propagation, crack initiation and corrosion.

Dwg.3/4

Abstract (Equivalent): US 5681404 A

A cladding tube having precipitates that vary in coarseness and density across the cladding wall, with coarse precipitates having an average diameter ranging from between about 0.15 and 2 micrometers in an inner region and fine precipitates having an average diameter ranging from between about 0.01 and 0.15 micrometers in an outer region, the cladding tube fabricated from a zirconium alloy tube by a method comprising steps of:

(a) performing a coarsening anneal on the zirconium alloy tube at a temperature of at least about 700 deg. C. for between about 1 and 100 hours such that precipitates coarsen throughout the entire tube; (b) selectively heat treating the outer region of the zirconium alloy tube by first heating the outer region to at least the alpha plus beta region while cooling the inner region of the tube and then rapidly quenching the outer region; and (c) performing one or more cold work steps on the zirconium alloy tube, each followed by an annealing step, the annealing step or steps being conducted at a temperature of greater than about 576 deg. C. , where the coarse precipitates in the inner region impart resistance to axial crack propagation in the cladding tube.

Dwg.3/3

US 5437747 A

Zirconium alloy cladding tube is fabricated by a method in which the tube is subjected to an anneal at a temp. above 621 deg.C, pref. at 775 deg.C for 4 hr. to coarsen pptes. throughout the tube. An outer region is selectively heated, pref. in an induction coil, to at least the alpha plug beta region, while the interior is cooled, and the outer region is then quenched. One or more cold walk steps are each followed by an anneal at a temp. greater than 576 deg.C. pref. at 620-650 deg.C.

USE/ADVANTAGE - Used in nuclear fuel elements. Tube is resistant to axial crack propagation.

(Dwg.0/4)

Title Terms: FABRICATE; ZIRCONIUM; ALLOY; CLAD; TUBE; PRODUCE; COMPONENT; HIGH; RESISTANCE; CRACK; PROPAGATE
Derwent Class: K05; M26; M29; X14
International Patent Class (Main): C22F-001/018; C22F-001/18; G21C-003/00
International Patent Class (Additional): G21C-003/06; G21C-003/07

35/7,DE/31 (Item 6 from file: 351)
DIALOG(R)File 351:Derwent WPI

009967240
WPI Acc No: 1994-234953/199428

Mfr. of nuclear fuel elements having fuel rods and cladding tubes - where the zirconium cladding tube has a zirconium@ alloy internal liner whose temp. during mfr. does not exceed the temp. when an incipient phase transformation to beta phase takes place.

Patent Assignee: ABB ATOM AB (ALLM)
Inventor: DAHLBAECK M; DAHLBACK M
Number of Countries: 020 Number of Patents: 010
Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week	
WO 9415343	A1	19940707	WO 93SE1070	A	19931215	199428	B
FI 9502981	A	19950616	WO 93SE1070	A	19931215	199538	
			FI 952981	A	19950616		
EP 674800	A1	19951004	WO 93SE1070	A	19931215	199544	
			EP 94903202	A	19931215		

JP 8505225	W	19960604	WO 93SE1070	A	19931215	199648
			JP 94515076	A	19931215	
EP 674800	B1	19970326	WO 93SE1070	A	19931215	199717
			EP 94903202	A	19931215	
US 5620536	A	19970415	WO 93SE1070	A	19931215	199721
			US 94284648	A	19940811	
DE 69309305	E	19970430	DE 609305	A	19931215	199723
			WO 93SE1070	A	19931215	
			EP 94903202	A	19931215	
ES 2102810	T3	19970801	EP 94903202	A	19931215	199737
SE 506174	C2	19971117	SE 923871	A	19921218	199801
JP 3031714	B2	20000410	WO 93SE1070	A	19931215	200023
			JP 94515076	A	19931215	

Priority Applications (No Type Date): SE 923871 A 19921218
Cited Patents: EP 121204; EP 155603; EP 194797; EP 425465; SE 459340
Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
WO 9415343	A1	E	21	G21C-021/02	
Designated States (National): FI JP US					
Designated States (Regional): AT BE CH DE DK ES FR GB GR IE IT LU MC NL PT SE					
JP 3031714	B2		7	G21C-003/20	Previous Publ. patent JP 8505225 Based on patent WO 9415343
EP 674800	A1	E		G21C-021/02	Based on patent WO 9415343
Designated States (Regional): CH DE ES LI SE					
JP 8505225	W		21	G21C-003/20	Based on patent WO 9415343
EP 674800	B1	E	8	G21C-021/02	Based on patent WO 9415343
Designated States (Regional): CH DE ES LI SE					
US 5620536	A		5	G21C-021/02	Based on patent WO 9415343
DE 69309305	E			G21C-021/02	Based on patent EP 674800 Based on patent WO 9415343
ES 2102810	T3			G21C-021/02	Based on patent EP 674800
FI 9502981	A			G21C-000/00	
SE 506174	C2			G21C-003/20	

Abstract (Basic): WO 9415343 A

Method of mfr. of a nuclear fuel element comprising a composite cladding tube with a Zr or Zr alloy inner component suitable for PCI resistant composite cladding. The composite cladding has an outer Zr alloy component which constitutes the supporting part of the composite cladding tube, e.g. Zircalloy 2, Zircalloy 4, or Zr-2.5Nb. The cladding tube is mfd. by fabricating an ingot of the compsn. of the inner component and an ingot of the compsn. of the outer component, respectively. The ingots are separately machined into a billet of a suitable dimension. The ingots are joined together and extruded to a tube blank. The blank is further machined by cold rolling and intermediate heat treatment operations, and a final heat treatment in the final dimensions.

The mfr. of the inner component from the ingot prodn. up to completion of the cladding tube, comprises forging, rolling, extrusion, heat treatment and final heat treatment.

USE/ADVANTAGE - Method of mfr. of nuclear fuel elements comprising fuel rods whose cladding tubes are provided with Zr or Zr alloy internal liner. Nuclear fuel elements has improved resistance to the corrosive effect of water and steam in case of damage.

Dwg.1/1

Abstract (Equivalent): EP 674800 B

A method for manufacturing a nuclear fuel element comprising a composite cladding tube with an inner component of zirconium or a

zirconium alloy suitable as inner component in a PCI-resistant composite cladding as well as an outer component of a zirconium alloy intended to constitute the supporting part of a composite cladding tube, such as, for example, Zircaloy 2, zircaloy 4 or Zr 2.5 Nb, wherein the cladding tube is manufactured by fabricating an ingot of the composition of the inner component and an ingot of the composition of the outer component, respectively, and machining them separately into a billet of a suitable dimension and thereafter joining them together and extruding to a tube blank and machining it further by means of cold rolling and intermediate heat-treatment operations and a final heat treatment in the final dimension, characterised in that the inner component of zirconium or a zirconium alloy during the manufacture, from the production of an ingot up to the completion of a cladding tube, comprising forging, rolling, extrusion, heat treatment and final heat treatment, is only subjected to heat influence at temperatures in the alpha-phase range below the temperature when an incipient beta-phase transformation takes place.

Dwg.0/1

Abstract (Equivalent): US 5620536 A

A method of manufacturing a composite cladding tube of a nuclear fuel element which is resistant to pellet-clad interaction, the composite cladding tube comprising an inner portion formed from a first component selected from the group consisting of zirconium and a zirconium alloy and an outer portion formed from a second component selected from the group consisting of Zircaloy 2, Zircaloy 4 and Zr 2.5 Nb, the method comprising the steps of: (a) providing an ingot of the first component, (b) forging rolling, extruding, and heat treating the ingot of the first component to form an inner billet, (c) positioning the inner billet from step (b) within an outer machined billet formed from an ingot of the second component, (d) extruding the joined billets from step (c) to form a joined tube blank, and (e) machining the tube blank from step (d) to provide the composite cladding tube, (f) the steps (b)-(e) being conducted at a temperature below that which causes incipient beta-phase transformation within the first component.

Dwg.0/1

Title Terms: MANUFACTURE; NUCLEAR; FUEL; ELEMENT; FUEL; ROD; CLAD; TUBE; ZIRCONIUM; CLAD; TUBE; ZIRCONIUM; ALLOY; INTERNAL; LINING; TEMPERATURE; MANUFACTURE; TEMPERATURE; INCIPIENT; PHASE; TRANSFORM; BETA; PHASE; PLACE
Derwent Class: K05; M21; M26
International Patent Class (Main): G21C-000/00; G21C-003/20; G21C-021/02
International Patent Class (Additional): C21D-009/08; C22F-001/00; C22F-001/18; G21C-003/06; G21C-003/07

35/7,DE/32 (Item 7 from file: 351)
DIALOG(R)File 351:Derwent WPI

009375448
WPI Acc No: 1993-068926/199309
Annealing of zirconium@ based alloy to improve nodular corrosion resistance - giving prod. used in the cladding of fuel elements in nuclear reactors
Patent Assignee: GENERAL ELECTRIC CO (GENE)
Inventor: TAYLOR D F
Number of Countries: 010 Number of Patents: 005
Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week	
EP 529907	A1	19930303	EP 92307522	A	19920818	199309	B
US 5188676	A	19930223	US 91749052	A	19910823	199310	

TW 198733	A	19930121	TW 92101397	A	19920225	199326
JP 5209257	A	19930820	JP 92220556	A	19920820	199338
JP 2677933	B2	19971117	JP 92220556	A	19920820	199751

Priority Applications (No Type Date): US 91749052 A 19910823
Cited Patents: GB 2086945; US 4000013; US 4351678
Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
EP 529907	A1	E		C22F-001/02	
Designated States (Regional): CH DE ES FR IT LI SE					
US 5188676	A		7	C22F-001/00	
JP 5209257	A		6	C22F-001/18	
JP 2677933	B2		6	C22F-001/18	Previous Publ. patent JP 5209257
TW 198733	A			C21D-001/26	

Abstract (Basic): EP 529907 A

A process for annealing a zirconium based alloy member having a cold worked or beta quenched crystal structure comprises: annealing the member in an atmos. consisting of oxygen and the balance an inert atmos. to form an adherent black oxide on the member. Also claimed is a process for recrystallisation annealing a zirconium based alloy as above.

Pref. the atmos. comprises at least 0.1 vol.% of oxygen, 0.1 grams of oxygen per square metre surface area of zirconium based alloy, less than 20 ppm of nitrogen, less than 2 ppm of hydrogen and less than 10 ppm of water.

USE/ADVANTAGE - Used in the cladding of fuel elements in nuclear reactors. The reductions in nodular corrosion resistance found in the annealed zirconium alloy member is maintained or improved.

Dwg.1/5

Abstract (Equivalent): US 5188676 A

Annealing zircalloy part comprises annealing to form an adherent black oxide, in atmos. at effective amt. of oxygen to form the black oxide, and the remainder an inert atmos.

Recrystallising annealing the zircalloy part comprises formation of adherent black oxide, in atmos. of O2 and inert gases.

Atmos. e.g. comprises less than 20 pts. per million of N2, less than 2 pts. per million H2 and less than 10 ppm water.

USE/ADVANTAGE - To anneal zircalloy part with cold worked or beta quenched crystal structure, to mitigate redn. modular corrosion resistance due to recrystallisation annealing.

Dwg.0/5

Title Terms: ANNEAL; ZIRCONIUM; BASED; ALLOY; IMPROVE; NODULE; CORROSION; RESISTANCE; PRODUCT; CLAD; FUEL; ELEMENT; NUCLEAR; REACTOR
Derwent Class: K05; M29; X14
International Patent Class (Main): C21D-001/26; C22F-001/00; C22F-001/02; C22F-001/18
International Patent Class (Additional): C22C-016/00

35/7,DE/33 (Item 8 from file: 351)
DIALOG(R)File 351:Derwent WPI

009142826
WPI Acc No: 1992-270264/199233
Nuclear fuel element component of zirconium@ alloy - with high temp. corrosion resistance to cooling water, fuel and fission products
Patent Assignee: SIEMENS AG (SIEI)
Inventor: STEINBERG E

Number of Countries: 006 Number of Patents: 007
Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week	
EP 498259	A2	19920812	EP 92101295	A	19920127	199233	B
US 5245645	A	19930914	US 91745904	A	19910816	199338	
			US 92839629	A	19920221		
EP 498259	A3	19930616	EP 92101295	A	19920127	199405	
US 5296058	A	19940322	US 91745904	A	19910816	199411	
			US 92839629	A	19920221		
			US 9338671	A	19930326		
EP 498259	B1	19960327	EP 92101295	A	19920127	199617	
DE 59205799	G	19960502	DE 505799	A	19920127	199623	
			EP 92101295	A	19920127		
ES 2084847	T3	19960516	EP 92101295	A	19920127	199627	

Priority Applications (No Type Date): EP 91112979 A 19910801; EP 91101463 A 19910204
Cited Patents: No-SR.Pub; EP 171675; EP 196286; EP 405172; EP 85553; FR 1327734; FR 2509510; US 4963323
Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
EP 498259	A2	G	10	G21C-003/07	
Designated States (Regional): DE ES FR GB SE					
US 5245645	A		9	G21C-003/06	CIP of application US 91745904
US 5296058	A		10	C22C-016/00	Cont of application US 91745904
					Div ex application US 92839629
					Div ex patent US 5245645
EP 498259	B1	G	12	G21C-003/07	
Designated States (Regional): DE ES FR GB SE					
DE 59205799	G			G21C-003/07	Based on patent EP 498259
ES 2084847	T3			G21C-003/07	Based on patent EP 498259
EP 498259	A3			G21C-003/07	

Abstract (Basic): EP 498259 A

A structural part for a nuclear fuel element, esp. a cladding tube or a spacer, consists of a Zr alloy contg. (by wt.) 0.8-1.7 % Sn, at least one of 0.07-0.5% Fe, 0.05-0.35% Cr and up to 0.1% Ni, 700-2000 ppm. O and/or up to 150 ppm. Si, balance Zr and impurities. Secondary phases of Fe, Cr and Ni pptd. in the matrix have a geometrical mean diameter of 0.1-0.3 microns and the alloy is in the max. 10% recrystallised state, a specimen of the alloy having an average grain size of max. 3 microns when recrystallisation annealed to a 95-99% recrystallised state.

Also claimed are (i) processes for producing the structural parts; and (ii) structural parts in which the alloy is Zircaloy-2 or -4.

ADVANTAGE - The structural part has high corrosion resistance w.r.t. the fuel and fission products and w.r.t. cooling water even when used at relatively high temp., e.g. the temp. which prevails in a PWR and which is higher than that in a BWR

Dwg.0/3

Abstract (Equivalent): EP 498259 B

A structural part for a nuclear fuel element, esp. a cladding tube or a spacer, consists of a Zr alloy contg. (by wt.) 0.8-1.7 % Sn, at least one of 0.07-0.5% Fe, 0.05-0.35% Cr and up to 0.1% Ni, 700-2000 ppm. O and/or up to 150 ppm. Si, balance Zr and impurities. Secondary phases of Fe, Cr and Ni pptd. in the matrix have a geometrical mean diameter of 0.1-0.3 microns and the alloy is in the max. 10% recrystallised state, a specimen of the alloy having an average grain size of max. 3 microns when recrystallisation annealed to a 95-99% recrystallised state.

Also claimed are (i) processes for producing the structural parts;

and (ii) structural parts in which the alloy is Zircaloy-2 or -4.
ADVANTAGE - The structural part has high corrosion resistance w.r.t. the fuel and fission products and w.r.t. cooling water even when used at relatively high temp., e.g. the temp. which prevails in a PWR and which is higher than that in a BWR
EP-498259 Structural part for a reactor fuel element, in particular a cladding tube for a fuel rod filled with nuclear fuel or spacers for such fuel rods, having the following features: (a) the material of the structural part is a zirconium alloy with at least one alloy component of the group oxygen and silicon, with tin as alloy component, having at least one alloy component of the group iron, chromium and nickel and with a remainder consisting of zirconium and unavoidable contaminants, (b) in the zirconium alloy there is selected a content of oxygen in the range of 700 to 2000 ppm, a content of silicon of up to 150 ppm, a content of iron in the range of 0.07 to 0.5% by weight, a content of chromium in the range of 0.05 to 0.35% by weight, a content of nickel of up to 0.1% by weight and a content of tin in the range of 0.8 to 1.7% by weight, (c) the geometric mean value of the diameter of the alloy components of the group iron, chromium and nickel, precipitated in the matrix of the zirconium alloy as secondary phases, is selected in the range of 0.1 to 0.3 micro-m, and (d) the degree of recrystallisation of the zirconium alloy is less than or equal to 10% and a sample of the zirconium alloy has after a recrystallisation annealing with a degree of recrystallisation of 97+/-2% a geometric mean value of the grain diameter which is smaller than or equal to 3 micro-m.
(Dwg.0/3)

Abstract (Equivalent): US 5245645 A
The structural part comprises a Zr alloy contg. 0.07-0.5 wt.% Fe, 0.05-0.35 wt.% Cr, upto 0.1 wt.% Ni and 0.8-1.7 wt.% Sn, with 700-2000 ppm O and upto 150 ppm Si. The Fe, Cr and Ni are pptd. out of a matrix of the Zr alloy as sec. phases, having a dia. with a geometric average value of 0.1-0.3 microns. The deg. of recrystallisation of the Zr alloy is upto 10%. A sample of the alloy after a recrystallisation annealing has a deg. of recrystallisation of 97 +/- 2% and has a grain size with a geometric mean value upto 3 microns. The alloy pref. has a texture with a Kearns parameter fr = 0.6 to 0.8. Ratio of Fe to Cr is pref. 2:1.
USE/ADVANTAGE - Used for a cladding or casing tube for a fuel rod or spacer. High corrosion resistance both to the nuclear fuel and fission prods. but also to the water coolant, even at relatively high temps. (Dwg.0/3)
US5296058 The structural part is produced by (d) annealing a Zr alloy in the beta range below the m.pt to dissolve pptd. out alloy ingredients, then quenching at least 30 deg.C./s, at a temp. transition through the alpha+beta range in which both hexagonal and bcc structures are present, (b) annealing at a 1st temp. in the alpha range until formation of pptes. of the alloy ingredients having a ppte. dia. with a geometric mean value of 0.1-0.3 microns, (c) hot forging at a 2nd temp. in the alpha range below the 1st temp., (d) hot rolling or hot extending at a temp. in the alpha range below the 1st. temp., and (e) cold rolling in at least two steps, including recrystallisation annealing between the rolling steps with a recrystallisation deg. of 95-99% at a temp. in the alpha range, while cold pilgering in at least two steps sepd. by recrystallisation annealing.
USE/ADVANTAGE - Used as cladding or casing for a nuclear fuel rod or spacer. Optimal corrosion resistance to water at elevated temps.
(Dwg.0/3)

Title Terms: NUCLEAR; FUEL; ELEMENT; COMPONENT; ZIRCONIUM; ALLOY; HIGH; TEMPERATURE; CORROSION; RESISTANCE; COOLING; WATER; FUEL; FISSION;

PRODUCT
Derwent Class: K05; M26
International Patent Class (Main): C22C-016/00; G21C-003/06; G21C-003/07
International Patent Class (Additional): C22F-001/18

35/7,DE/34 (Item 9 from file: 351)
DIALOG(R)File 351:Derwent WPI

004776906
WPI Acc No: 1986-280247/198643
Thin zirconium-niobium alloy tubing prodn. - useful for nuclear fuel cladding
Patent Assignee: WESTINGHOUSE ELECTRIC CORP (WESE)
Inventor: MCDONALD S G; SABOL G P
Number of Countries: 007 Number of Patents: 005
Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week	
EP 198570	A	19861022	EP 86300259	A	19860116	198643	B
JP 61210166	A	19860918	JP 8611802	A	19860122	198644	
ES 8708021	A	19871116	ES 551049	A	19860120	198751	
EP 198570	B	19900829				199035	
KR 9309986	B1	19931013	KR 86376	A	19860122	199438	

Priority Applications (No Type Date): US 85693546 A 19850122
Cited Patents: A3...8741; EP 71193; EP 85553; FR 2576322; LU 41401; No-SR.Pub; US 2894866; US 3341373; US 3865635
Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
EP 198570	A	E	19		
Designated States (Regional): BE FR GB IT					
EP 198570	B				
Designated States (Regional): BE FR GB IT					
KR 9309986	B1			C22F-001/18	

Abstract (Basic): EP 198570 B
Thin walled tubing is made from a Zr-Nb alloy, contg. 1-2.5 wt.% Nb as homogeneously dispersed finely divided particles, by (i) beta-treating a billet of the alloy, (ii) extruding at max. 650 deg.C to form a tube shell, (iii) multistage cold working with intermediate anneals at below 650 deg.C, and (iv) final annealing at below 600 deg.C. The resulting tubing has a microstructure contg. a homogeneous dispersion of Nb particles of less than 800 angstroms size.
USE/ADVANTAGE - The process is useful for prodn. of nuclear fuel cladding. Tubing, of 0.040 in. or less wall thickness and excellent corrosion resistance, is made without requiring extensively long final annealing times. (19pp Dwg.No.0/5)
Abstract (Equivalent): EP 198570 B
A process for fabricating thin-walled tubing having a wall thickness of about 1 mm or less from a zirconium-niobium alloy containing from 1 to 2.5 per cent by weight niobium as homogeneously dispersed finely divided particles and optionally up to 0.5 per cent by weight of copper, iron, molybdenum, nickel, tungsten, vanadium or chromium as a third element, balance, apart from impurities, zirconium, characterised by beta-treating a zirconium niobium alloy billet containing from 1 to 2.5 per cent by weight niobium; extruding said beta-treated billet at a temperature no higher than 650 deg C to form a tube shell; further deforming said tube shell by cold working the same in a plurality of cold working stages; annealing said tube shell,

between each of said stages of cold working, at a temperature below 650 deg C, and final annealing the resultant tubing at a temperature below 600 deg C, so as to produce a microstructure of the material having niobium particles of a size below about 800 angstroms (80nm) homogeneously dispersed therein.
Title Terms: THIN; ZIRCONIUM; NIOBIUM; ALLOY; TUBE; PRODUCE; USEFUL; NUCLEAR; FUEL; CLAD
Derwent Class: K05; M26; M29; P51
International Patent Class (Main): C22F-001/18
International Patent Class (Additional): B21C-023/01; B21C-029/00; B21C-037/06

35/7,DE/35 (Item 10 from file: 351)
DIALOG(R)File 351:Derwent WPI

004756367
WPI Acc No: 1986-259708/198640
Mfr. of nuclear fuel cladding tubes of zirconium alloy - using annealing temp. and time sufficient to equilibrate pptd. sec. phase particles to improve corrosion resistance
Patent Assignee: SANTRADE LTD (SANV)
Inventor: ANDERSSON E T; SCHEMEL J H; WILSON S A
Number of Countries: 008 Number of Patents: 009
Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week	
EP 196286	A	19861001	EP 86850081	A	19860306	198640	B
SE 8501216	A	19860913				198645	
SE 8501217	A	19860913				198645	
JP 61270360	A	19861129	JP 8651584	A	19860311	198702	
EP 196286	B	19890517				198920	
DE 3663372	G	19890622				198926	
US 4908071	A	19900313	US 88226517	A	19880729	199016	
KR 9309987	B1	19931013	KR 861739	A	19860311	199438	
JP 2583488	B2	19970219	JP 8651584	A	19860311	199712	

Priority Applications (No Type Date): SE 851217 A 19850320; SE 851216 A 19850320
Cited Patents: FR 2219978; GB 2079317; US 4450016
Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
EP 196286	A	E	6	P	
Designated States (Regional): DE FR GB IT					
EP 196286	B	E		P	
Designated States (Regional): DE FR GB IT					
JP 2583488	B2		5	P	Previous Publ. patent JP 61270360
SE 8501216	A			P	
SE 8501217	A			P	
JP 61270360	A			P	
DE 3663372	G			P	
US 4908071	A			P	
KR 9309987	B1			P	

Abstract (Basic): EP 196286 A
Mfr. nuclear fuel cladding tubes of zirconium alloy contg. 1-5 wt.% alloying elements such as Sn, Fe, Cr and Ni to improve the corrosion resistance in a water-cooled reactor environment is claimed. Method comprises annealing the material, after extrusion and/or between cold rollings, at 625-790 deg.C in the alpha phase range and at a

combination of temp. and time which gives complete equilibrium between Zr matrix and the precipitated secondary phase particles. An annealing parameter (A) is used, defined by formula (I) where t = annealing time (hours), R = gas constant (cal/mole.degree), T = temp. (K), and 65000 is the activation energy (cal/mole).

The value of this parameter (A) is kept above a critical value of 2.3 x 10 power -14, so that min. annealing times at various temps. are as follows: 0.5 h at 790 deg.C; 3.9 h at 725 deg.C; 22.2 h at 675 deg.C; 56.5 h at 650 deg.C; and 151.3 h at 625 deg.C.

ADVANTAGE - Equilibration of the precipitated particles ensures a minimum dissolved Fe content in the Zr matrix, thus improving corrosion resistance.

Dwg.0/0

EP 196286 B

Mfr. nuclear fuel cladding tubes of zirconium alloy contg. 1-5 wt.% alloying elements such as Sn, Fe, Cr and Ni to improve the corrosion resistance in a water-cooled reactor environment is claimed. Method comprises annealing the material, after extrusion and/or between cold rollings, at 625-790 deg.C in the alpha phase range and at a combination of temp. and time which gives complete equilibrium between Zr matrix and the precipitated secondary phase particles. An annealing parameter (A) is used, defined by formula (I) where t = annealing time (hours), R = gas constant (cal/mole.degree), T = temp. (K), and 65000 is the activation energy (cal/mole).

The value of this parameter (A) is kept above a critical value of 2.3 x 10 power -14, so that min. annealing times at various temps. are as follows: 0.5 h at 790 deg.C; 3.9 h at 725 deg.C; 22.2 h at 675 deg.C; 56.5 h at 650 deg.C; and 151.3 h at 625 deg.C.

ADVANTAGE - Equilibration of the precipitated particles ensures a minimum dissolved Fe content in the Zr matrix, thus improving corrosion resistance. (6pp Dwg.No.0/0)

Abstract (Equivalent): EP 196286 B

Method of making cladding tubes of a zirconium alloy containing 1-5 percent by weight of alloying elements including S, Fe, Cr and Ni and the rest essentially Zr, for the purpose of improving the corrosion resistance to general corrosion in media typical of water cooled thermal nuclear reactors at high pressure and high temperature, at which the material is annealed after extrusion and/or between cold rollings in the alpha-phase range at a temperature, within the interval 625-790 deg.C, characterised in that there is used a combination of temperature and time which essentially gives complete equilibrium between zirconium matrix and the precipitated secondary phase particles at which the combination of temperature/time is defined by an annealing parameter where t is the annealing time in hours, T is the absolute temperature, R is the general gas constant in cal/mole - degree, which at equilibrium shall exceed a critical value Ac = 2.3, 10-14, where said avlue means the following examples of shortest annealing times within the temperature interval 625-790 deg.C. Annealing temp. (deg.C), annealing time (hours) 790, 0.5; 725, 3.9; 675, 22.2; 650, 56.5; 625, 151.3.

Abstract (Equivalent): US 4908071 A

Cladding tubes of Zr alloy contg. 1-5 wt.% alloying elements including Sn, Fe, Cr and Ni, for use in nuclear reactors, are mfd. from extruded material which is annealed after extrusion and between cold rollings in the alpha phase range at 625-790 deg.C. Combination of temp. and time gives equilibrium between Zr matrix and pptd. second phase particles. Annealing temp. and time are defined by a parameter A = t.e power (-6500/RT), where t is annealing time, T is absolute temp., and is R is the gas constant. Tubes are cooled at 3 deg.C/min. max.

ADVANTAGE - Improved corrosion resistance for longer service times.

(4pp)
Title Terms: MANUFACTURE; NUCLEAR; FUEL; CLAD; TUBE; ZIRCONIUM; ALLOY; ANNEAL; TEMPERATURE; TIME; SUFFICIENT; EQUILIBRIUM; PRECIPITATION; SEC; PHASE; PARTICLE; IMPROVE; CORROSION; RESISTANCE
Derwent Class: K05; M29
International Patent Class (Main): C22F-001/18
International Patent Class (Additional): G21C-003/06; G21C-003/20

35/7,DE/36 (Item 11 from file: 351)
DIALOG(R)File 351:Derwent WPI

004722285
WPI Acc No: 1986-225627/198635
Zirconium(alloy) or titanium (alloy) seamless tube mfr. - by thermo-mechanical treatment of welded tube
Patent Assignee: WESTINGHOUSE ELECTRIC CORP (WESE)
Inventor: BARRY R F; SABOL G P
Number of Countries: 004 Number of Patents: 004
Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week	
BE 904221	A	19860812	BE 904221	A	19860212	198635	B
JP 61186462	A	19860820	JP 8628067	A	19860213	198640	
FR 2580524	A	19861024	FR 861951	A	19860213	198649	
US 4690716	A	19870901	US 85701326	A	19850213	198737	

Priority Applications (No Type Date): US 85701326 A 19850213
Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
BE 904221	A		14		

Abstract (Basic): BE 904221 A
Seamless tube is mfd. from a welded starting tube of zirconium (alloy) or titanium (alloy), having a heterogeneous structure as a result of welding, by (i) through-heating of successive axial segments of the welded tube, including the weld, for uniform transformation to the beta phase and (ii) quenching of the tube segments, steps (i) and (ii) being affected rapidly to avoid beta phase grain growth beyond 200 microns diameter. Finally (iii) the cooled tube is rapidly deformed to final shape.
USE/ADVANTAGE - Resulting tubes are useful for nuclear fuel cams, aviation hydraulic piping, heat exchanger tubes, and condenser tubes. It has a uniform alpha phase structure with improved mechanical properties and corrosion resistance. (14pp Dwg.No.0/0)
Abstract (Equivalent): US 4690716 A
Seamless tubes of zirconium or titanium alloy are formed by welding together a precursor tube to give a heterogeneous phase.
The tube is then theated in successive stages to convert all the material, including the weld zone, into beta phase material before quenching sufficiently rapidly to give a fire grained beta structure with no grain above 200 microns. Finally the quenched tube is subject to cold reduction until the final size and shape is reached.
ADVANTAGE - The process produces seamless tube suitable e.g. as nuclear fuel rod cladding, at reduced cost. (4pp)
Title Terms: ZIRCONIUM; ALLOY; TITANIUM; ALLOY; SEAM; TUBE; MANUFACTURE; THERMO; MECHANICAL; TREAT; WELD; TUBE
Derwent Class: K05; M21; P51
International Patent Class (Additional): B21B-003/02; B21B-021/00; C21B-000/00; C21D-008/10; C22F-001/18

35/7,DE/37 (Item 12 from file: 351)
DIALOG(R)File 351:Derwent WPI

003740419
WPI Acc No: 1983-736618/198333
Fabricating zirconium alloys to improve steam corrosion resistance - by reducing temp. of working and annealing during fabrication
Patent Assignee: WESTINGHOUSE ELECTRIC CORP (WESE)
Inventor: MCDONALD S G; SABOL G P
Number of Countries: 009 Number of Patents: 007
Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week	
EP 85553	A	19830810	EP 83300455	A	19830128	198333	B
JP 58204144	A	19831128	JP 8313618	A	19830128	198402	
ES 8602148	A	19860301	ES 519378	A	19830128	198619	
US 4584030	A	19860422	US 84571122	A	19840113	198619	
CA 1214978	A	19861209				198702	
EP 85553	B	19881123				198847	
DE 3378537	G	19881229				198902	

Priority Applications (No Type Date): US 82343787 A 19820129; US 84571122 A 19840113
Cited Patents: BE 691169; FR 1415082; No-SR.Pub; US 3431104; US 3567522; US 4000013; US 3645800
Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
EP 85553	A	E	19		
Designated States (Regional): DE FR GB IT SE					
EP 85553	B	E			
Designated States (Regional): DE FR GB IT SE					

Abstract (Basic): EP 85553 A
Zircalloy article has a microstructure adjacent the surface which comprises a random distribution of precipitates (below 1100 microns) and the surface after 5 days exposure to 454 deg.C, 10.3 MPa steam has an adherent oxide film. The Zircalloy article is fabricated by conventional beta treatment of a billet followed by thermomechanical working below 625 deg.C. The working may be cold working with intermediate anneals at 500-600 deg.C. Used as cladding for nuclear components in pressurised water and boiling water reactors. Improved high temp. steam corrosion resistance is obtd.
Abstract (Equivalent): EP 85553 B
A process for fabricating Zircaloy alloy shapes comprising the steps of heating a Zircaloy intermediate product to an elevated temperature above the alpha+beta to beta transus temperature and quenching said Zircaloy intermediate product from said elevated temperature to a temperature below the alpha+beta to alpha transus temperature to form precipitates having an average diameter below about 1100 Angstroms, then extruding said alloy at a temperature between about 500 and 600 deg.C, then cold working said alloy in series of cold pilgering steps each of which is preceded by a thermal treatment step comprising essentially of only a low temperature anneal, said low temperature anneal limited to about 500 to 600 deg.C, and after the final cold pilgering step subjecting the resulting material to a final anneal at about 466 to 600 deg.C. (11pp)
Abstract (Equivalent): US 4584030 A
Zircalloy workpiece has a microstructure region adjacent to a

surface which comprises a random distribution of pptes. The pptes have an average size of less than 1100 angstroms and the surface is exposed for 5 days to steam at a pressure of 10.3 MPa and temp of 454 deg.C to form a uniform, continuous, adherent oxide film.

Pref. the microstructure comprises a stress relieved cold worked structure, pref comprising polygonal alpha grains and an anisotropic crystallographic structure.

ADVANTAGE - Good long term corrosion resistances in a high temp steam environment. (11pp)

Title Terms: FABRICATE; ZIRCONIUM; ALLOY; IMPROVE; STEAM; CORROSION; RESISTANCE; REDUCE; TEMPERATURE; WORK; ANNEAL; FABRICATE
Derwent Class: K05; M29; X14
International Patent Class (Additional): C22C-016/00; C22F-001/18; G21C-003/06

35/7,DE/38 (Item 13 from file: 351)
DIALOG(R)File 351:Derwent WPI

003715170
WPI Acc No: 1983-711353/198328
Nuclear fuel element cladding mfr. - by cold working and heat treating zirconium alloy tube lined with zirconium to recrystallise the latter
Patent Assignee: GENERAL ELECTRIC CO (GENE)
Inventor: DAVIES J H; ROSENBAUM U S
Number of Countries: 001 Number of Patents: 001
Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week	
US 4390497	A	19830628				198328	B

Priority Applications (No Type Date): US 81304011 A 19810921; US 7945225 A 19790604
Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
US 4390497	A		8		

Abstract (Basic): US 4390497 A

Method is claimed for making nuclear fuel element cladding tube from a tube shell consisting of Zr alloy contg. more than 5000 ppm impurities, with a protective barrier bonded to the inside of the tube shell and consisting of Zr contg. less than 5000 ppm impurities with thickness 1-30% of that of the composite tube.

The method comprises (i) reducing the dia. of the tube shell and lining layer by cold working to obtain the desired dia. and wall thickness; (ii) heat treating between each reduction step to fully recrystallise the Zr alloy; and (iii) finally heat treating to complete recrystallisation of the Zr metal layer to produce a fine-grained microstructure and stress-relieves but does not fully recrystallise the Zr alloy.

The heat treatment enables the Zr metal layer to be recrystallised without causing undesirable grain growth. The microstructure of the protective barrier layer can be further improved by shot peening. The Zr alloy retains elongated grain structure and has higher strength at high strain rates.

Title Terms: NUCLEAR; FUEL; ELEMENT; CLAD; MANUFACTURE; COLD; WORK; HEAT; TREAT; ZIRCONIUM; ALLOY; TUBE; LINING; ZIRCONIUM; RECRYSTALLISATION; LATTER
Derwent Class: K05; M29; X14
International Patent Class (Additional): G21C-003/20

35/7,DE/39 (Item 14 from file: 351)
DIALOG(R)File 351:Derwent WPI

003425644
WPI Acc No: 1982-00021J/198247
Zirconium alloy nuclear fuel cladding tube prodn. - using controlled intermediate anneals during cold rolling
Patent Assignee: ASEA ATOM AB (ALLM)
Number of Countries: 006 Number of Patents: 006
Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week	
BE 893788	A	19821103				198247	B
DE 3224685	A	19830120				198304	
SE 8104214	A	19830207				198308	
FR 2509510	A	19830114				198309	
JP 58025466	A	19830215				198312	
FI 8202395	A	19830228				198315	

Priority Applications (No Type Date): SE 814214 A 19810707
Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
BE 893788	A		13		

Abstract (Basic): BE 893788 A
Zirconium alloy nuclear fuel cladding tubes are produced by (i) extrusion at below 680 deg.C; (ii) cold rolling with intermediate anneals, at least one of which is carried out at above 650 deg.C in the alpha phase region and is followed by cooling at at least 5 deg.C/min. from the annealing temp. to 650 deg.C, the intermediate anneals after this at least one anneal being carried out at max. 600 deg.C; and (iii) final annealing.

The resulting tubes contain a homogeneous distribution of very fine second phase particles giving good nodular corrosion resistance and good mechanical properties.
Title Terms: ZIRCONIUM; ALLOY; NUCLEAR; FUEL; CLAD; TUBE; PRODUCE; CONTROL; INTERMEDIATE; ANNEAL; COLD; ROLL
Derwent Class: K05; M21; M29; P51; P54; X14
International Patent Class (Additional): B21C-023/00; B23B-000/00; C22C-016/00; C22F-001/00; G21C-003/06; G21C-021/00

35/7,DE/40 (Item 1 from file: 32)
DIALOG(R)File 32:METADEX(R)

0905565 MA Number: 199209-61-0661
A Method of Manufacturing Cladding Tubes for Fuel Rods for Nuclear Reactors.
Andersson, T
Sandvik
Patent: EP0425465, European Patent 26 Oct. 1990
Auszuge aus den Europaischen Patentanmeldungen, Teil I 7, (18), 1722 2 May 1991
Country of Publication: Germany
Journal Announcement: 9209
Document Type: Patent
Language: ENGLISH

Abstract: A method of making cladding tubes for fuel rods of a Zr alloy containing (in wt.%) 1.2-1.7 wt.% Sn, 0.07-0.24 Fe, 0.05-0.15 Cr, 0-0.08 Ni and the balance essentially Zr and ordinary impurities for the purpose of improving the resistance to nodular corrosion under operating conditions in boiling water nuclear reactors with or without a barrier of Zr having a Sn addition of 0-0.5 bonded to its inside surface includes the steps of: (a) extruding a perforated billet of the Zr-based alloy within the alpha -phase range for obtaining a tube, (b) cold rolling the tube with multiple cold working passes to the size for the final cold rolling and annealing the tube at a temperature in the alpha -phase range subsequent to each cold rolling pass, (c) heating an outer portion of the tube wall to the beta -phase range for a time sufficient to transform the outer portion of the tube wall to beta -phase while cooling the inner portion of the tube wall at a temperature sufficiently low that essentially no metallurgical changes occur at the inner portion of the tube wall and then cooling the tube sufficiently rapidly to transform the beta -phase into a structure of alpha -grains with a fine distribution of intermetallic particles in the alpha -grain boundaries and thereby improve the resistance to nodular corrosion; (d) cold rolling the thus partially beta -quenched tube to the final cladding tube size; (e) annealing the cold rolled tube at a temperature of 400-650 deg C.

Descriptors: Patent; Zirconium base alloys-- Claddings; Nuclear fuel elements; Tubemaking; Cold rolling; Heat treatment; Corrosion resistance-- Microstructural effects; Microstructure-- Cooling effects

WEST
EAST

S nuclear$1 near3 fuel near3 clad$3
L1 + (zirconi$3 or zircaloy$3)
Lots of good art in Japio Derwent

DERWENT-ACC-NO: 1999-266204
DERWENT-WEEK: 200051

TITLE: Zirconium alloy nuclear fuel cladding production

INVENTOR: ISOBE, T; SUDA, Y

PATENT-ASSIGNEE: MITSUBISHI MATERIALS CORP[MITV]

PRIORITY-DATA: 1998JP-0287800 (October 9, 1998) , 1997JP-0278935 (October 13, 1997)

PATENT-FAMILY:

PUB-NO	PUB-DATE	LANGUAGE	PAGES	MAIN-IPC
US 6125161 A	September 26, 2000	N/A	000	G21C 003/07
FR 2769637 A1	April 16, 1999	N/A	039	C21D 008/00
JP 11194189 A	July 21, 1999	N/A	024	G21C 003/06

APPLICATION-DATA:

PUB-NO	APPL-DESCRIPTOR	APPL-NO	APPL-DATE
US 6125161A	Div ex	1998US-0169968	October 13, 1998
US 6125161A	N/A	1999US-0397094	September 16, 1999
FR 2769637A1	N/A	1998FR-0012784	October 13, 1998
JP 11194189A	N/A	1998JP-0287800	October 9, 1998

INT-CL_(IPC): C21D001/26; C21D008/00 ; C22C016/00 ; C22F001/00 ; C22F001/18 ; G21C003/06 ; G21C003/07

ABSTRACTED-PUB-NO: FR 2769637A

BASIC-ABSTRACT: NOVELTY - In the production of nuclear fuel cladding of a zirconium alloy containing Nb or Nb+Ta, annealing is carried out at 550-850

deg. C for 1-4 h such that the log of the cumulative anneal parameter is -20 to -15 and satisfies a mathematical relationship relating it to the Nb or Nb+Ta content.

DETAILED DESCRIPTION - Nuclear fuel cladding is produced by subjecting a zirconium alloy of composition (by wt.) 0.2-1.7% Sn, 0.18-0.6% Fe, 0.07-0.4% Cr, 0.05-1.0% Nb, optionally 0.01-0.1% Ta, balance zirconium and impurities, including at most 60 ppm N, to hot forging, solution heat treatment, hot extrusion, repeated annealing and cold rolling, and final stress relief annealing, the annealing being carried out at 550-850 deg. C for 1-4 h such that the cumulative anneal parameter approx. SAi (where approx. SAi = approx. Sti asterisk exp(-40000/Ti)) satisfies the relationships of log approx. SAi = -20 to -15 and log approx. SAi = -18-10XNb to -15-3.75(XNb-0.2), in which Ai = anneal parameter for the 'i'th anneal, ti = anneal duration (h) for the 'i'th anneal, Ti = the anneal temperature (K) for the 'i'th anneal and XNb = the Nb and optional Ta content (in wt.%). An INDEPENDENT CLAIM is also included for a zirconium alloy nuclear fuel cladding made by the above process.

USE - For producing nuclear fuel cladding tubes for a PWR.

ADVANTAGE - The annealing conditions provide the cladding tube with better corrosion resistance and creep properties than conventional cladding tubes and thus has a long useful life.

ABSTRACTED-PUB-NO: US 6125161A
EQUIVALENT-ABSTRACTS: NOVELTY - In the production of nuclear fuel cladding of a zirconium alloy containing Nb or Nb+Ta, annealing is carried out at 550-850 deg. C for 1-4 h such that the log of the cumulative

anneal parameter is -20 to -15 and satisfies a mathematical relationship relating it to the Nb or Nb+Ta content.

DETAILED DESCRIPTION - Nuclear fuel cladding is produced by subjecting a zirconium alloy of composition (by wt.) 0.2-1.7% Sn, 0.18-0.6% Fe, 0.07-0.4% Cr, 0.05-1.0% Nb, optionally 0.01-0.1% Ta, balance zirconium and impurities, including at most 60 ppm N, to hot forging, solution heat treatment, hot extrusion, repeated annealing and cold rolling, and final stress relief annealing, the annealing being carried out at 550-850 deg. C for 1-4 h such that the cumulative anneal parameter approx. SAi (where approx. SAi = approx. Sti asterisk exp(-40000/Ti)) satisfies the relationships of log approx. SAi = -20 to -15 and log approx. SAi = -18-10XNb to -15-3.75(XNb-0.2)', in which Ai = anneal parameter for the 'i'th anneal, ti = anneal duration (h) for the 'i'th anneal, Ti = the anneal temperature (K) for the 'i'th anneal and XNb = the Nb and optional Ta content (in wt.%). An INDEPENDENT CLAIM is also included for a zirconium alloy nuclear fuel cladding made by the above process.

USE - For producing nuclear fuel cladding tubes for a PWR.

ADVANTAGE - The annealing conditions provide the cladding tube with better corrosion resistance and creep properties than conventional cladding tubes and thus has a long useful life.

CHOSEN-DRAWING: Dwg.0/0

TITLE-TERMS:
ZIRCONIUM ALLOY NUCLEAR FUEL CLAD PRODUCE

DERWENT-CLASS: K05 M26 M29 X14

Nuclear Reactors

CPI-CODES: K05-B04B; M26-B06; M26-B06C; M26-B06J; M26-B06N;
M26-B06T; ~~M29~~-C01;

non-ferrous alloys.

EPI-CODES: X14-B04; X14-B04A;

SECONDARY-ACC-NO:
CPI Secondary Accession Numbers: C1999-078658
Non-CPI Secondary Accession Numbers: N1999-198505

> to search in Derwent

KO5-B04B.cpi. or

KO5$.cpi. ← truncate.

Harry,

I obtained the first results in this printout (L61 and L62) from searching the claim language and some process conditions. However, I was unhappy with those results, so I searched on Zr alloy AND nuclear fuel cladding AND manufacture/fabricate... AND pipe... I was happier with these results, they are listed at the end of the printout.

John

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=> d his nofile

     FILE 'REGISTRY' ENTERED AT 10:47:54 ON 12 JUL 2002
L1          55532 SEA ZR/ELS AND AYS/CI
L2          53412 SEA ZR AND AYS/CI
L3          45373 SEA L1 AND 50-100/MAC

     FILE 'HCAPLUS' ENTERED AT 10:48:53 ON 12 JUL 2002
L4        1297849 SEA ALPHA?
L5          59969 SEA L4(3A)(PHASE? OR STRUCTUR? OR MICROSTRUCTUR?)
L6         178683 SEA MICROSTRUCTUR?
L7                QUE PRODUC? OR PROD# OR GENERAT? OR MANUF? OR MFR# OR CREAT?
                  OR FORM## OR FORMING# OR FORMAT? OR MAKE# OR MADE# OR MAKING#
                  OR FABRICAT? OR PREPAR? OR PREP#
L8        1084859 SEA CREEP? OR STRESS? OR STRAIN? OR DEFORMAT? OR FATIGUE? OR
                  FRACTUR? OR EMBRITTLE?
L9         176174 SEA L8(4A)(RESIST? OR RECOVER? OR STRENGTH?) OR TOUGHNESS? OR
                  RESILEN?
L10        737678 SEA NUCLEAR?
L11        193290 SEA L10(4A)FUEL? OR URANIUM? OR PLUTONIUM OR PU
L12          5494 SEA L10(3A)FUEL?(3A)CLAD?
L13               QUE CLAD? OR CASING? OR SHEATH? OR ENSHEATH? OR ENCAS? OR
                  ENCAPSULAT? OR ENVELOP? OR OVERLAID? OR LAMIN? OR LAMEL? OR
                  ENCAS? OR WRAP? OR SURROUND?
L14        695326 SEA LATH? OR STRIP? OR SWATH# OR BAND## OR SLAT### OR ROW###
L15        694636 SEA TUBE#  OR TUBING# OR TUBUL? OR TUBIFORM? OR PIPE# OR
                  PIPING# OR PIPELI? OR CONDUIT? OR CYLIND?
L16         27688 SEA (FAST## OR SWIFT## OR RAPID? OR QUICK?)(4A)(COOL?)
L17         25940 SEA (COLD# OR METAL?)(2A)(WORK? OR METALWORK?)
L18        250121 SEA ANNEAL? OR RECRYSTALLI?
L19         19467 SEA ZIRCALOY? OR (ZIRONI## OR ZR)(3A)(ALLOY? OR AMALGAM? OR
                  MIXTUR? OR ADMIX? OR BLEND?)
L20         46961 SEA L3
L21         57035 SEA L19 OR L20 OR ZIRCALOY(2W)4
L22          7517 SEA ACICULAR
L23          4269 SEA NEEDLE?(3A)(LIKE#)
L24         11713 SEA L22 OR L23
L25               QUE HEAT? OR WARM? OR HOT# OR CALEFACT? OR TORREFACT? OR
                  PYROL? OR THERMOL? OR TEPEFACT? OR MELT? OR DISSOL? OR FUSE#
                  OR FUSING# OR FUSION? OR (HIGH## OR HEIGHTEN OR ELEVAT?)(2A)(TE
                  MP# OR TEMPERATUR?)

L26        946440 SEA 70/SC,SX OR 71/SC,SX
L27        599536 SEA 56/SX,SC
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L28            QUE BETA
L29            QUE (BINARY OR DUAL OR TWO)(3A)PHASE?
L30      30738 SEA L21 AND L7
L31       2577 SEA L30 AND L4
L32       1144 SEA L30 AND L5
L34         66 SEA L32 AND (L14 OR LATH?)
L35         34 SEA L34 AND L8
L36          7 SEA L34 AND L9
L37        130 SEA L31 AND L14
L38         56 SEA L37 AND L8
L39         13 SEA L37 AND L9
L40          1 SEA L39 AND L10
L41       2802 SEA L21 AND L12
L42        292 SEA L41 AND L4
L43        120 SEA L41 AND L5
L44        190 SEA L42 AND L7
L45         71 SEA L43 AND L7
L46        190 SEA L44 AND L4
L47         71 SEA L45 AND L5
L48         71 SEA L44 AND L5
L49         71 SEA L45 AND L4
L50         80 SEA L46 AND L8
L51          4 SEA L46 AND L9
L52         71 SEA L47 OR L48 OR L49
L53         38 SEA L52 AND L8
L54          4 SEA L52 AND L9
L55        293 SEA L41 AND L18
L56         54 SEA L55 AND L17
L57          2 SEA L56 AND L16
L58         60 SEA L55 AND L4
L59         39 SEA L58 AND L28
L60         31 SEA L59 AND L7
L61          1 SEA L60 AND L14
L62          2 SEA L60 AND L9
L63         15 SEA L36 OR L40 OR L51 OR L54 OR L57 OR L61 OR L62
L64          5 SEA L39 NOT L63
L65       2951 SEA L30 AND L15
L66        561 SEA L65 AND L12
L67         42 SEA L66 AND L9
L68          2 SEA L67 AND L4

=> d L63 1-15 cbib abs hitind hitrn
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L63 ANSWER 1 OF 15 HCAPLUS COPYRIGHT 2002 ACS

2002:94998 Document No. 136:174386 Degradation of the mechanical properties of **Zircaloy-4** due to hydrogen embrittlement.
Bertolino, G.; Meyer, G.; Ipina, J. Perez (Centro Atomico Bariloche and Instituto Balseiro, Bariloche, Argent.). Journal of Alloys and Compounds, 330-332, 408-413 (English) 2002. CODEN: JALCEU. ISSN: 0925-8388. Publisher: Elsevier Science S.A..

AB During nuclear reactor operation, the embrittlement of components **made** of **Zr**-based **alloys** is obsd. The degrdn. of their mech. properties is due to the combined effect of H absorption and the damage caused by n irradn. The authors studied the influence of H content on the fracture **toughness** of a **Zircaloy-4** alloy. Compact tension (CT) specimens were obtained from a hot-rolled, annealed and finally cold-rolled material. The obsd. **microstructure** consisted of .**alpha**.-Zr rounded grains with diams. of .apprx.15 .mu.m. Selection of the tested material was guided by the need to perform expts. on samples with a texture equiv. to

the cladding components of Candu-type nuclear reactors. The specimens were fatigue pre-cracked and H charged before testing. Two different reactions were performed. Specimens with a final H content ranging from 10 to 400 ppm were obtained by electrochem. charging and those with a final concn. of up to 2000 ppm were charged by absorption under a gaseous atm. In both cases, an homogeneous distribution of dissolved H and hydride phases was obtained. The dependence of the **toughness** on temp. and H content was measured on CT specimens. The anal. was performed in terms of J-integral and resistance curves.

CC 71-12 (Nuclear Technology)

ST **Zircaloy** mech property hydrogen embrittlement nuclear reactor

IT Nuclear reactors
(during nuclear reactor operation, embrittlement of components **made** of **Zr**-based **alloys** is obsd.)

IT Alloys, properties
RL: PRP (Properties)
(during nuclear reactor operation, embrittlement of components **made** of **Zr**-based **alloys** is obsd.)

IT Fracture **toughness**
(influence of H content on fracture **toughness** of a **Zircaloy-4** alloy.)

IT **Microstructure**
(**microstructure** consisted of .**alpha**.-Zr rounded grains)

IT **Nuclear fuel** element **cladding**
(samples with texture equiv. to fuel cladding components of Candu-type nuclear reactors.)

IT **11068-95-4**
RL: PRP (Properties)
(Degrdn. of mech. properties of)

IT 1333-74-0, Hydrogen, processes
RL: PEP (Physical, engineering or chemical process); PROC (Process)
(embrittlement; degrdn. of mech. properties of **Zircaloy-4** due to hydrogen embrittlement)

IT **11068-95-4**
RL: PRP (Properties)
(Degrdn. of mech. properties of)

L63 ANSWER 2 OF 15 HCAPLUS COPYRIGHT 2002 ACS

2000:478927 Document No. 133:153873 Influence of microstructure on fretting fatigue behavior of a near-alpha titanium alloy. Satoh, Toyoichi (Jet Engine Division, 3rd Research Center, Technical Research & Development Institute, Japan Defense Agency, Tachikawa, 190-8533, Japan). ASTM Special Technical Publication, STP 1367(Fretting Fatigue), 295-307 (English) 2000. CODEN: ASTTA8. ISSN: 0066-0558. Publisher: ASTM.

AB To investigate the effect of microstructure on the fretting fatigue behavior of a near-.alpha. titanium alloy, fretting fatigue tests were carried out using DAT54, which is used in compressor blades and disks in aircraft gas turbine engines. Two kinds of microstructure in DAT54 were **prepd**. using different soln. heat treatment temps.: one is the equiaxed .**alpha**. + .**alpha**. **lath** **microstructure** and the other is the transformed .beta. structure. The plain and fretting **fatigue strengths** for the equiaxed .**alpha**. + .**alpha**. **lath** **microstructure** are higher than for the transformed .beta. structure. Fretting reduced **fatigue strengths** by a factor of three for both materials. Sensitivity to microstructure in fretting fatigue is relatively low compared with plain fatigue. Shot peening improved fretting fatigue life, because of lower tangential force between the specimen and the contact pad and because of residual stress in

compression induced by shot peening treatment.
CC 56-10 (Nonferrous Metals and Alloys)
IT **180254-61-9**, DAT54
RL: PEP (Physical, engineering or chemical process); PRP (Properties); TEM (Technical or engineered material use); PROC (Process); USES (Uses)
(effect of microstructure on fretting fatigue behavior of near-alpha titanium alloy)
IT **180254-61-9**, DAT54
RL: PEP (Physical, engineering or chemical process); PRP (Properties); TEM (Technical or engineered material use); PROC (Process); USES (Uses)
(effect of microstructure on fretting fatigue behavior of near-alpha titanium alloy)

L63 ANSWER 3 OF 15 HCAPLUS COPYRIGHT 2002 ACS
2000:242474 Document No. 133:10033 Influence of cladding microstructure on the low enthalpy failures in RIA simulation tests. Garde, A. M. (ABB Combustion Engineering Nuclear Fuel, Windsor, CT, 06095, USA). ASTM Special Technical Publication, STP 1354(Zirconium in the Nuclear Industry: Twelfth International Symposium, 1998), 234-255 (English) 2000. CODEN: ASTTA8. ISSN: 0066-0558. Publisher: ASTM.
AB A review with 30 refs. Welding used during the **prepn**. of specimens for Reactivity Initiated Accident (RIA) simulation testing from fuel rods previously irradiated in reactors introduces the following 4 microstructural changes in the specimen: (a) **annealing** of irradn. damage, (b) a change from an .**alpha**.-**phase structure** of **Zircaloy** to a transformed .**beta**. structure in the cladding, (c) dissoln. of the hydride rim **formed** under the oxide on the cladding tube outer surface during normal irradn. and possible radial-oriented hydride repptn. at the transformed .**beta**. platelet boundaries, and (d) repptn. of 2nd-phase particles previously dissolved due to radiation damage. A 5th factor, the change in the texture of **Zircaloy**, is also introduced due to the welding operation. The possible effect of these 5 changes on the specimen fracture **toughness** and failure enthalpy is evaluated. The published data on the mech. properties of irradiated and unirradiated transformed **beta** structure of **Zircaloy** charged with H are reviewed to evaluate the impact of the 5 anticipated microstructural changes on the failure enthalpy. The available RIA simulation test fracture data with low-failure enthalpy are reviewed. The limited failure path information available appears to indicate that microstructural factors have contributed to the low-enthalpy failures. The applicability of the results from the low-enthalpy RIA test failures to Light Water Reactor (LWR) nuclear fuel should be based on the representativeness of the RIA specimen microstructure to that of the LWR fuel cladding.
CC 71-0 (Nuclear Technology)
Section cross-reference(s): 56
IT Fracture **toughness**
Light-water nuclear reactors
Nuclear fuel element **cladding**
Nuclear reactor accident
Simulation and Modeling, physicochemical
Structural phase transition
Welding of metals
(influence of cladding microstructure on low enthalpy failures in RIA simulation tests)
IT **11068-94-3**, **Zircaloy**-2 **11068-95-4**, **Zircaloy-4**
RL: PRP (Properties)
(influence of cladding microstructure on low enthalpy failures in RIA simulation tests)

IT **11068-94-3**, **Zircaloy**-2 **11068-95-4**,
Zircaloy-4
RL: PRP (Properties)
(influence of cladding microstructure on low enthalpy failures in RIA simulation tests)

L63 ANSWER 4 OF 15 HCAPLUS COPYRIGHT 2002 ACS
1998:246006 Document No. 128:311490 Micromechanisms of fatigue crack propagation in Ti3Al based alloys. Wu, X.; Bowen, P. (School of Metallurgy and Materials, The University of Birmingham, UK). Materials Science and Technology, 14(3), 206-216 (English) 1998. CODEN: MSCTEP. ISSN: 0267-0836. Publisher: Institute of Materials.

AB Direct monitoring of the influence of the .**alpha**.2 and .beta. **phases** on a growing crack was performed using an FEG SEM for Ti-23Al-9Nb-2Mo-1Zr-1.2Si (at.-%) and Ti-23Al-11Nb-0.9Si (at.-%) Ti3Al based alloys. Crack growth rates are faster across individual .alpha.2 **laths** than across .beta. **laths** and/or along .alpha.2/.beta. **lath** interfaces. Fatigue cracks propagate incrementally through the .**alpha**.2 **phase** by decohesion of a favored slip **band** rather than crossing it catastrophically in one cycle. The **formation** of intersecting slip **bands** can lead to a tortuous crack path and a decreased av. crack growth rate in the .**alpha**.2 **phase.** When a crack meets the .beta. phase, the most common phenomenon obsd. is crack deflection. The fatigue crack then extends continuously along .alpha.2/.beta. interfaces under the effect of a mixed mode local stress intensity factor range. The basket weave microstructure achieves the max. **fatigue** crack growth **resistance** from .alpha.2/.beta. interfaces. Bridging and blunting can reduce fatigue crack growth rate remarkably. However, crack bridging happens only with larger .beta. **laths** and blunting is mainly seen only for secondary cracks. The efficiency of bridging and blunting thus depends on the ratio of load bearing capability of the .beta. **laths** involved over the local effective .DELTA.K range. Mechanisms operating during fatigue crack propagation are also compared with those obsd. during monotonic fracture.

CC 56-12 (Nonferrous Metals and Alloys)

IT 120474-22-8, Aluminum 23 niobium 11 silicon 0.9 titanium 65.1 atomic
206645-94-5, Aluminum 23 molybdenum 2 niobium 9 silicon 1.2 titanium 63.8 zirconium 1 atomic
RL: PRP (Properties)
(fatigue crack propagation in Ti3Al based alloys)

IT **206645-94-5**, Aluminum 23 molybdenum 2 niobium 9 silicon 1.2 titanium 63.8 zirconium 1 atomic
RL: PRP (Properties)
(fatigue crack propagation in Ti3Al based alloys)

L63 ANSWER 5 OF 15 HCAPLUS COPYRIGHT 2002 ACS
1997:127704 Document No. 126:189567 Damage maps of titanium alloys. Helbert, A. L.; Feaugas, X.; Clavel, M. (Div. Mecanique, Univ. Technol. Compiegne, Fr.). Revue de Metallurgie/Cahiers d'Informations Techniques, 93(12), 1539-1549 (French) 1996. CODEN: CITMDA. ISSN: 0035-1563. Publisher: Revue de Metallurgie.

AB Damage evolution was studied on four .alpha./.beta. titanium alloys using the local approach to fracture. Tensile tests on notched and smooth specimens have been performed to fracture or interrupted before fracture. Besides, finite element calcns. were conducted for each specimen design to provide the mech. parameters in the bulk of the specimen during loading up to fracture. In order to quantify damage, midsections of the specimens were polished and etched. In each surface element, void d. and length were quantified and linked to the mech. parameters. Void nucleation has

been studied and then void nucleation and growth kinetics have been examd. A nucleation criterion of voids at the .alpha./.beta. interface, based on plastic strain and hydrostatic stress, has been identified for each whereas voids are **created** in the **.alpha.-phase** for a const. plastic strain. Void no. and length increase exponentially with resp. strain and triaxiality. They both drastically increase for crit. couples of strain and triaxiality. Three types of fracture occur depending on strain and triaxiality. The first, assisted by triaxiality, leads to fracture by void growth. The second appears for less triaxiality and under a certain amt. of strain. In this case, the quantity of voids is the cause of failure. At last, when the stress triaxiality is too low, no voids are **created** during loading but plastic strain localizes in shearing **bands** and the .alpha.-grains rotate so as to accommodate more plastic strain. The materials then develop a plastic strain instability at a macroscopic scale. These types of fracture were reported on fracture maps for the different alloys studied. The internal stress (X) is directly linked to the triaxiality level needed for void nucleation. So, this mech. parameter greatly influences the fracture modes experienced by the different materials. Besides, calcd. loading paths of points located ahead of the crack tip can be plotted on the fracture maps. Only a point located at a certain distance from the crack tip experiences enough strain and triaxiality to intersect the crit. growth curve. This distance from the crack corresponds to the exptl. zone where damage has been obsd. on an interrupted CT sample for a charge slightly before fracture. **Toughness** of the titanium alloys studied can be predicted as far as the fracture maps and loading paths ahead of the CT crack are detd.

CC 56-12 (Nonferrous Metals and Alloys)

IT 12743-70-3, TA6V **52293-96-6,** Ti6-2-4-6 185402-05-5, TD5AC
RL: PEP (Physical, engineering or chemical process); PRP (Properties); PROC (Process)
(fracture maps of titanium alloys)

IT **52293-96-6,** Ti6-2-4-6
RL: PEP (Physical, engineering or chemical process); PRP (Properties); PROC (Process)
(fracture maps of titanium alloys)

L63 ANSWER 6 OF 15 HCAPLUS COPYRIGHT 2002 ACS

1997:65744 Document No. 126:92972 The effect of hot working on the transformation microstructure of the titanium alloy Ti-17. Rowe, R. G.; Sundell, R. E.; Gigliotti, M. F. X. (GE Corporate Research and Development, Schenectady, NY, 12309, USA). Titanium '95: Science and Technology, Proceedings of the World Conference on Titanium, 8th, Birmingham, UK, Oct. 22-26, 1995, Meeting Date 1995, Volume 3, 2250-2257. Editor(s): Blenkinsop, P. A.; Evans, W. J.; Flower, H. M. Institute of Materials: London, UK. (English) 1996. CODEN: 63XBAB.

AB The microstructure of beta forged, soln. heat treated, and aged Ti-17 consists of coarse Widmanstaetten alpha plates in a matrix of "aged beta" within prior beta grains. This microstructure has a good balance of properties such as **creep** and **fatigue fracture resistance.** However, Ti-17, thermomechanically processed to thin sections, cools rapidly, and a fine **.alpha.**+.beta. **lath structure** with low fracture **toughness** is **formed**. The control of cooling rate after forging offers a means of **producing** coarse **lath** microstructures without re-soln. heat treatment. Investigation of cooling rate effects allowed an opportunity to det. if prior deformation affects the CCT diagram of Ti-17. It was found that there is an affect of forging on the high temp. transformation regime but that the effect on microstructure is diminished at lower temps.

CC 56-8 (Nonferrous Metals and Alloys)
IT **37329-07-0**, Ti-17
RL: PEP (Physical, engineering or chemical process); PRP (Properties); PROC (Process)
(effect of hot working on transformation microstructure of Ti alloy Ti-17)
IT **37329-07-0**, Ti-17
RL: PEP (Physical, engineering or chemical process); PRP (Properties); PROC (Process)
(effect of hot working on transformation microstructure of Ti alloy Ti-17)

L63 ANSWER 7 OF 15 HCAPLUS COPYRIGHT 2002 ACS
1996:74892 Document No. 124:124120 Creep behavior of cast TiAl based intermetallics. Kim, S.; Cho, W.; Hong, C.-P. (Dep. Metallurgical Engineering, Yonsei Univ., Seoul, S. Korea). Mater. Sci. Technol., 11(11), 1147-55 (English) 1995. CODEN: MSCTEP. ISSN: 0267-0836.
AB Const. load tensile creep tests were carried out on the cast TiAl based intermetallics Ti-47Al-2Mn, Ti-47Al-2Zr, and Ti-48Al (at.-%), **prepd**. by plasma arc melting. Two microstructural conditions dependent on heat treatment were evaluated as follows: a fully lamellar (FL) scheme consisting of a fully transformed coarse lamellar **structure** with **.alpha**.2 **lath** plus .gamma. **lath** within the grain interiors; and a duplex scheme consisting of fine equiaxed grains of .gamma. with .alpha.2/.gamma. lamellae. The steady state creep behavior of both microstructural conditions, for each compn., was studied under stresses of 70-300 MNm-2 in the temp. range 700-900.degree.C. The microstructure was found to have a pronounced influence on the **creep resistance**. The FL microstructure exhibited superior **creep resistance** to the duplex microstructure. At temps. and stress levels at which direct comparisons can be **made**, the steady state creep rates of the FL structures are an order of magnitude lower than those of the duplex structure. The apparent creep activation energies and stress exponents were measured for both microstructural conditions for each compn. The temp. and stress dependence of the steady state creep rate of both microstructures can be described by the power law creep equation, suggesting dislocation motion as the operative deformation mechanism.
CC 56-12 (Nonferrous Metals and Alloys)
IT **159222-73-8**, Aluminum 47, titanium 51, zirconium 2 (atomic)
RL: PRP (Properties)
(intermetallic; creep behavior of cast TiAl based intermetallics)
IT **159222-73-8**, Aluminum 47, titanium 51, zirconium 2 (atomic)
RL: PRP (Properties)
(intermetallic; creep behavior of cast TiAl based intermetallics)

L63 ANSWER 8 OF 15 HCAPLUS COPYRIGHT 2002 ACS
1991:518431 Document No. 115:118431 **Manufacture** of stainless steel **strip** having high strength and **toughness**. Takemoto, Toshihiko; Tanaka, Teruo; Murata, Yasushi (Nisshin Steel Co., Ltd., Japan). Jpn. Kokai Tokkyo Koho JP 02225647 A2 19900907 Heisei, 7 pp. (Japanese). CODEN: JKXXAF. APPLICATION: JP 1989-43237 19890227.
AB The stainless steel for cold-rolled **strip** contains C .ltoreq.0.15, Si 3.0-7.0, Mn .ltoreq.8.0, Ni 8.0-13.0, Cr 12.0-17.0, and N .ltoreq.0.10% with the Ni equiv. of preferably 8-14%. The cold-rolled **strip** has martensitic **.alpha.'-phase** **structure**, and is heat treated at 600-900.degree. to give the .gamma.-**.alpha.**' final **microstructure**. Thus, the hot-rolled **strip** (contg. C 0.036, Si 3.02, Mn 1.05, Ni 8.92, Cr 15.33, and N 0.021%) was heated for solid soln., cold rolled at 60%

draft/pass, and then reheated at 710.degree. for 2 min. The resulting **strip** showed tensile strength of 115 kg/mm2, vs. 83 kg/mm2 for the **strip** from SUS 304 stainless steeel.
IC ICM C22C038-00
ICS C21D009-46; C22C038-40
CC 55-11 (Ferrous Metals and Alloys)
ST stainless steel **strip** heat treatment; silicon stainless steel **strip**
IT 135951-85-8 135951-86-9 135951-87-0 135951-88-1 135951-89-2 135951-90-5 135951-91-6 135951-92-7 135951-93-8 **135953-12-7** 135953-13-8 135953-14-9
RL: USES (Uses)
(high-strength **strip** from, by cold rolling and heat treatment)
IT **135953-12-7**
RL: USES (Uses)
(high-strength **strip** from, by cold rolling and heat treatment)

L63 ANSWER 9 OF 15 HCAPLUS COPYRIGHT 2002 ACS
1990:61152 Document No. 112:61152 Effect of structure on the physicomechanical properties of zirconium + 2.5% niobium alloy sheets. Bryukhanov, A. A.; Tarasov, A. F.; Goncharov, A. B.; Nerodenko, M. M. (Odessa, USSR). Izv. Akad. Nauk SSSR, Met. (6), 161-4 (Russian) 1989. CODEN: IZNMAQ. ISSN: 0568-5303.
AB The mech. properties after **strip** rolling, heat treatment, and recrystn. were detd. for **Zr**-2.5% Nb **alloy**, and were related to the texture of **strip** specimens 2 mm thick. Anisotropy of the mech. properties was present after the **strip** **manuf**., and was retained after annealing in the **.alpha.-phase** range, but was removed by recrystn. annealing at 1000-1273 K in the .beta.-phase range. Impact **toughness** was decreased by cooling or quenching of recrystn. annealed specimens thus promoting the **formation** of an acicular martensitic phase.
CC 56-12 (Nonferrous Metals and Alloys)
ST zirconium niobium **strip** texture strength; recrystn zirconium niobium **toughness**
IT **50813-12-2**, Niobium 2.5, zirconium 98
RL: PRP (Properties)
(mech. properties of, **strip** texture effect on, annealing in relation to)
IT **50813-12-2**, Niobium 2.5, zirconium 98
RL: PRP (Properties)
(mech. properties of, **strip** texture effect on, annealing in relation to)

L63 ANSWER 10 OF 15 HCAPLUS COPYRIGHT 2002 ACS
1988:154707 Document No. 108:154707 Laser beam beta heat treatment of **Zircaloy**. Sabol, G. P.; McDonald, S. G.; Nurminen, J. I.; Jacobsen, W. A. (Westinghouse R and D Cent., Pittsburgh, PA, 15235, USA). ASTM Spec. Tech. Publ., 939(Zirconium Nucl. Ind.), 168-86 (English) 1987. CODEN: ASTTA8. ISSN: 0066-0558.
AB A com. viable process of .beta.-phase heat treatment of **Zircaloy** tubing is developed, that circumvents the problems of dimensional stability at the treatment temp., protection from oxidn., and control of .beta. grain size. The method utilizes a high-power, geometrically shaped laser beam that impinges on the surface of the tubing as the tubing is rotated and translated through the focused beam. Beam energy, surface temp., and translation speed are chosen such that the zone of the .beta.-phase formation penetrates .apprx.30% of the wall thickness, and

the self-quenching provided by the underlying material produces a uniformly **rapid cooling** rate. The laser treatment is performed in Ar-filled chamber on next-to-final-sized tubing, and tube properties are recovered by **cold working** to final size and final **annealing**. Tubing of **Zircaloy**-2 and **Zircaloy-4** produced by the laser beam .beta.-phase treatment is immune to nodules on both the inside and outside surfaces in the 24 h, 773 K steam test, and post-transition corrosion rates of **Zircaloy-4** are lower than those of conventionally processed tubing by factors of 2-3 at 633-673 K. The laser .beta. heat treatment process and the resultant tubing properties are described.

CC 56-5 (Nonferrous Metals and Alloys)
Section cross-reference(s): 73

IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, **Zircaloy**, laser .beta.-phase heat treatment of, corrosion resistance by)

IT **11068-94-3**, **Zircaloy** 2 **11068-95-4**, **Zircaloy 4**
RL: USES (Uses)
(laser-beam .beta.-phase heat treatment of, corrosion resistance by)

IT **11068-94-3**, **Zircaloy** 2 **11068-95-4**, **Zircaloy 4**
RL: USES (Uses)
(laser-beam .beta.-phase heat treatment of, corrosion resistance by)

L63 ANSWER 11 OF 15 HCAPLUS COPYRIGHT 2002 ACS
1985:619509 Document No. 103:219509 A transition stress in the creep of an **alpha-phase** zirconium alloy at high temperature. Donaldson, A. T.; Ecob, R. C. (Berkeley Nucl. Lab., CEGB, Berkeley/Gloucestershire, UK). Scr. Metall., 19(11), 1313-18 (English) 1985. CODEN: SCRMBU. ISSN: 0036-9748.

AB The strain-time curve **form** of PWR **Zircaloy-4** [**11068-95-4**] fuel cladding creep at 923-1073 K had a transition at a temp.-dependent stress, .sigma.T, under const. loads. The transition stress value increases with decreasing temp. and is the macroscopic yield stress. **Creep** occurs by a **recovery**-controlled dislocation mechanism above .sigma.T, but at a lower stress a grain-boundary diffusion mechanism dominates deformation.

CC 56-12 (Nonferrous Metals and Alloys)
Section cross-reference(s): 71

ST **Zircaloy** cladding creep transition stress

IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, **Zircaloy**, creep transition stress at high temp.)

IT **11068-95-4**
RL: USES (Uses)
(**nuclear** reactor **fuel cladding**, creep transition stress at high temp. in)

IT **11068-95-4**
RL: USES (Uses)
(**nuclear** reactor **fuel cladding**, creep transition stress at high temp. in)

L63 ANSWER 12 OF 15 HCAPLUS COPYRIGHT 2002 ACS
1982:604566 Document No. 97:204566 **Beta**-quenching of **Zircaloy** cladding tubes in intermediate or final size - methods to improve corrosion and mechanical properties. Andersson, T.; Vesterlund, G. (Sandvik AB, Sandviken, Swed.). ASTM Spec. Tech. Publ., 754(Zirconium Nucl. Ind.), 75-95 (English) 1982. CODEN: ASTTA8. ISSN: 0066-0558.

AB Three batches of **Zircaloy**-2 [**11068-94-3**] tubing were

.beta.-quenched prior to the final cold-rolling, cold rolled 80 %, and **annealed** at 475 - 575.degree.. A 4th batch was . **beta**.-quenched in the final size. For comparison, std. tubing was included in all tests performed. The 2nd-phase particles were studied by means of optical and SEM. Corrosion testing was carried out at 400.degree. and in high-temp. (475 - 500.degree.) high-pressure steam. The mech. tests comprised tension, burst, and creep testing under internal pressure. **.beta**.-Quenching instead of an intermediate or the final **anneal** results in significant structural changes. The most striking features are the **formation** of a structure consisting of plates of **.alpha.-phase** and the repptn. of much finer 2nd-phase particles in the plate boundaries. Cold-rolling of **.beta**.-quenched hollows followed by a final **anneal** in the .**alpha**.-range will give an equiaxed structure, but the size and distribution of the 2nd phase obtained in **.beta** .-quenching will not be markedly changed. The wt. gain at 400.degree. increases slightly as a result of **.beta**.-quenching in intermediate or final size. In high-pressure steam at 475 - 500.degree., on the other hand, such **.beta**.-quenching has a dramatic beneficial effect on the corrosion resistance. The short-term strength as measured in tension and burst testing is improved by **.beta** .-quenching of hollows or finished tubes, whereas such treatment results in a slight drop in ductility, esp. for tubing **.beta**.-quenched in the final size. The 400.degree. transverse **creep strength** is increased by the introduction of **.beta** .-quenching prior to the final cold-rolling. The improvement is caused mainly by small 2nd-phase particles, **formed** during **.beta** .-quenching, which gives rise to pptn. hardening.

CC 71-5 (Nuclear Technology)

ST **Zircaloy** cladding **beta** quenching; reactor fuel cladding corrosion prevention

IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, **Zircaloy**, .**beta**.-quenching of, for improved corrosion prevention and mech. properties)

IT **11068-94-3**
RL: PROC (Process)
(.**beta**.-quenching of cladding tubes of, for improved corrosion prevention and mech. properties)

IT **11068-94-3**
RL: PROC (Process)
(.**beta**.-quenching of cladding tubes of, for improved corrosion prevention and mech. properties)

L63 ANSWER 13 OF 15 HCAPLUS COPYRIGHT 2002 ACS

1978:178939 Document No. 88:178939 Nodular corrosion of the **Zircaloys**. Johnson, A. B., Jr.; Horton, R. M. (Corros. Res. Eng. Sect., Battelle Northwest, Richland, Wash., USA). ASTM Spec. Tech. Publ. (STP 633, Zirconium Nucl. Ind.), 295-311 (English) 1977. CODEN: ASTTA8. ISSN: 0066-0558.

AB Oxide nodules form on **Zircaloy** nuclear components under irradn. Similar nodules were obsd. on **Zircaloy** coupons in cold-rolled or extruded conditions after autoclave treatments at 475 and 500.degree. in steam at 1500-1700 psi. The stages of nodular corrosion in the autoclave were: nodule nucleation, growth, coalescence, propagation to accelerated uniform corrosion, and complete specimen oxidn. Observations on boiling water reactor (BWR) fuel rods suggest that a similar progressive attack has occurred; however, in no case has the in-reactor attack appeared to progress to the stage of complete component failure. Recent autoclave tests confirmed the nodular character of the attack on **cold-worked** materials. Alpha **anneals** (up to 790.degree.) did

not suppress the nodular attack consistently. However, alpha + beta (840.degree.) and beta (1010 and 1040.degree.) **anneals** did suppress the attack if they were followed by a **fast cool** . The efficacy of the **anneals** applied similarly to **Zircaloy**-2 [**11068-94-3**] and **Zircaloy-4** [**11068-95-4**]. Stresses assocd. with U-bend specimens and heavy (86 %) **cold work** did not enhance the nodular attack before stress relief occurred. The nodular attack on reactor components appears to depend on nuclear flux, and develops in oxygenated reactor coolants, principally in the vicinity of fuel rod spacers. Experience with irradiated specimens in reactor loops suggests that uniform concns. of dissolved O alone do not cause the large nodules which frequently develop on BWR fuel rods. Localized water chem. assocd. with flow disturbances or, in some cases, dissimilar metals in fuel spacers, may be factors in the nodular attack in-reactor.

CC 71-6 (Nuclear Technology)
Section cross-reference(s): 56

ST **Zircaloy** nodular corrosion reactor cladding; fuel cladding nodular corrosion

IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, **Zircaloy**, nodular corrosion of)

IT **11068-94-3 11068-95-4**
RL: PROC (Process)
(corrosion of reactor fuel claddings of, nodular)

IT **11068-94-3 11068-95-4**
RL: PROC (Process)
(corrosion of reactor fuel claddings of, nodular)

L63 ANSWER 14 OF 15 HCAPLUS COPYRIGHT 2002 ACS

1977:445639 Document No. 87:45639 The fatigue behavior of **.alpha** .-zirconium and **Zircaloy**-2 in the temperature range 20 to 700.degree.C. Snowden, K. U.; Stathers, P. A. (Mater. Div., Aust. At. Energy Comm. Res. Establ., Sutherland, Aust.). J. Nucl. Mater., 67(1-2), 215-28 (English) 1977. CODEN: JNUMAM.

AB Reverse plane-bending fatigue tests at 16 Hz were undertaken on Zr and **Zircaloy**-2 at temps. between 20 and 700.degree. to study the influence of temp., environment, and cold work on fatigue life and the nature of fatigue damage. The fatigue limit (for 106 cycles) at 20 and 300.degree. was 138.6 .+-. 24 and 107.4 .+-. 16 MN/m2 resp. for crystal bar Zr, and 235.0 .+-. 24 and 190.0 .+-. 10 MN/m2 resp. for **Zircaloy**-2. In addn., push/pull fatigue tests on **Zircaloy** -2 at 16 Hz showed that the fatigue limit (at 106 cycles) was 240 .+-. 20 MN/m2 at 20.degree. and 175 .+-. 20 MN/m2 at 300.degree. in good agreement with bending data. The temp. dependence of fatigue life at const. strain amplitude showed evidence of **fatigue strengthening** because of dynamic **strain** aging between 20 and 300.degree.. Above 300.degree., fatigue life decreased with increasing temp. Prior cold working by amts. up to 15% **strain** was detrimental to **fatigue resistance.** For **Zircaloy**-2 at 20.degree., cold work reduced fatigue limit by .apprx.30%. Examn. of the microstructure of fatigue specimens revealed that the fatigue crack path was transcryst. at low temps. and intercyrst. at high temps. The transition from a transcryst. to an intercryst. mode of failure occurred at .apprx.500.degree. for crystal bar Zr and .apprx.600.degree. for **Zircaloy**-2. The low-temp. mode of deformation was characterized by slip-**band** clustering, twinning, and slip-**band** extrusion. The high-temp. mode of deformation was characterized by fine slip, grain-boundary migration, and **formation** of a diamond grain structure.

CC 71-6 (Nuclear Technology)

Section cross-reference(s): 56, 75

ST fatigue zirconium temp; **Zircaloy** 2 fatigue temp; crystal transition zirconium fatigue temp; reactor fatigue zirconium; fuel fatigue zirconium

IT **Nuclear** reactor fuels and fuel elements
(fatigue behavior of zirconium and **Zircaloy** 2 in relation to)

IT 7440-67-7, properties **11068-94-3**
RL: PRP (Properties)
(fatigue of, temp. effect on)

IT **11068-94-3**
RL: PRP (Properties)
(fatigue of, temp. effect on)

L63 ANSWER 15 OF 15 HCAPLUS COPYRIGHT 2002 ACS

1968:438267 Document No. 69:38267 Diffusion bonding in vanadium and zirconium. Vermani, S. K.; Murarka, S. P.; Agarwala, R. P. (Chem. Div., Bhabha At. Res. Centre, Bombay, India). India At. Energy Comm., Bhabha At. Res. Centre, BARC-303 6 pp. (English) 1967. CODEN: IABRAA.

AB Investigations were carried out to study the possibility of using V as a protective coating over Zr since the H **formed** due to pyrolytic and radiolytic decompn. of org. coolants in nuclear reactors reacts with the cladding material, **Zr** and **Zr alloys**, and seriously affects the mech. properties. Even at low pressures, complete hydriding of Zr is possible. The clad can, however, be protected by coating it with a metal inert to H and having suitable nuclear and mech. properties. If such a coating be given, the problem of its compatibility with clad becomes important. Nuclear pure Zr, and V rods of 1/4-in. diam. were machined and abraded on carborundum and emery papers followed by electropolishing. Zr was electropolished in glacial HOAc and HClO4 bath, and V in a 10% H2SO4 bath. The polished ends of V and Zr samples were placed in contact with each other in a die locked by screwing the threaded cap. Some couples were **prepd.** by **annealing** to give diffusion bonding for nearly 10 hrs. at 400.degree. and for an hr. at 950.degree. in a purified He atm. The couples thus **formed** were further **annealed** in a controlled furnace at 400-1050.degree. in a purified He atm. at a pressure of 0.5 .times. 10-3 mm. Hg. Air-quenched couples were sectioned perpendicular to the diffusion front and diffusion **bands** were measured with an accuracy of 10 .mu. on a microscope. At <800.degree., very little diffusion was observed in the **.alpha**.-phase of Zr. A photomicrograph of V-Zr couple **annealed** at 800.degree. for 34 days shows a narrow diffusion **band**, while the couple **annealed** at <800.degree. for longer periods did not show any diffusion **band** at all. In Zr at .gtoreq.900.degree., only one diffusion **band** was observed. It appears from the phase diagram of V-Zr system that 2 phases are possible. One is a compd. of definite compn. ZrV2, and other is .delta.-phase. The phase appearing in these couples is ZrV2 and the absence of .delta.-phase can be attributed to the difficulty in distinguishing the phase under a microscope as this phase might have more or less a structure similar to that of V owing to high V content. The lack of appreciable diffusion in **.alpha**.-Zr is attributed to several reasons. There is no solid soly. of V in **.alpha**.-Zr while in the **.beta**.-phase, its increases with temp. to 1230.degree.. The low soly. of V in **.alpha**.-Zr is one of the controlling factors resulting in low diffusivity. The phase change of Zr from h.c.p. lattice in **.alpha**.-phase to b.c.c. lattice in the **.beta**.-phase increases the self-diffusion parameter. To what extent it affects is difficult to evaluate. No measurements were **made** to test the strength of the bond but the bonding appeared to be quite firm and can be explained by the fact that V and Zr **form** the terminal solid soln. The temp. of operation of a reactor is nearly

400.degree., at that temp. there will only be little, if any, interpenetration between V and Zr. No visible diffusion **band** appeared when a couple was **annealed** for 56 days at 400.degree.. At least from the compatibility point of view, V is a suitable barrier to protect Zr from the H embrittlement in org.-cooled reactors. 13 references.

CC 56 (Nonferrous Metals and Alloys)

IT **Nuclear** reactor **fuels**, uses and miscellaneous
(**claddings**, diffusion coating of, for prevention of hydrogen embrittlement)

IT Coating materials
(vanadium, on zirconium **nuclear** reactor **fuel** **cladding** materials, embrittlement by hydrogen in relation to)

IT 1333-74-0, properties
RL: PRP (Properties)
(embrittlement by, of zirconium **cladding** materials for **nuclear** reactor **fuels**, coating with vanadium for prevention of)

=> d L64 1-5 cbib abs hitind hitrn

L64 ANSWER 1 OF 5 HCAPLUS COPYRIGHT 2002 ACS

1997:389629 Document No. 127:98419 Effects of microstructure on the fracture **toughness** of Ti3Al-based titanium aluminides. Wu, X.; Bowen, P. (Sch. Metall. Mater., Univ. Birmingham, Birmingham, B15 2TT, UK). Metallurgical and Materials Transactions A: Physical Metallurgy and Materials Science, 28A(6), 1357-1365 (English) 1997. CODEN: MMTAEB. ISSN: 1073-5623. Publisher: Minerals, Metals & Materials Society.

AB The influence of microstructure on the fracture **toughness** of Ti-23Al-9Nb-2Mo-1Zr-1.2Si and Ti-23Al-11Nb-0.9Si (at.%) Ti3Al-based alloys has been investigated. Basket-weave microstructures comprising different vol. fractions of **.alpha**.2 and retained .beta. phases were **produced** by systematic heat treatments. Besides the vol. fraction of the retained .beta.-phase, the av. size of the .beta.**-laths** has also been used to characterize these microstructures. The **toughness** of both alloys was examd. at room temp. The brittle transgranular fracture modes were controlled by microstructure. However, the **toughness** is not detd. solely by the vol. fraction retained .beta.-phase, and a linear relationship has been obtained between the fracture **toughness** and av. size of the retained .beta.- **laths**. The **toughness** of the alloys at room temp. is controlled primarily by the width of retained .beta.**-laths** rather than by the retained .beta.-vol. fraction.

CC 56-12 (Nonferrous Metals and Alloys)

ST titanium aluminide microstructure fracture **toughness**

IT Fracture **toughness**
(effects of microstructure on fracture **toughness** of Ti3Al-based titanium aluminides)

IT 186003-49-6, Ti-23Al-11Nb-0.9Si **191978-05-9**
RL: PRP (Properties)
(effects of microstructure on fracture **toughness** of Ti3Al-based titanium aluminides)

IT **191978-05-9**
RL: PRP (Properties)
(effects of microstructure on fracture **toughness** of Ti3Al-based titanium aluminides)

L64 ANSWER 2 OF 5 HCAPLUS COPYRIGHT 2002 ACS

1992:412651 Document No. 117:12651 Mechanical properties of powder

metallurgy Ti-829 and Ti-25Al-10Nb-3V-1Mo **produced** by rapid omnidirectional compaction. Osborne, N. R.; Porter, W. J.; Eylon, D. (Res. Inst., Univ. Dayton, Dayton, OH, USA). SAMPE Q., 22(4), 21-8 (English) 1991. CODEN: SAMQA2. ISSN: 0036-0821.

AB Two powder-metallurgy high-temp. Ti alloys were compared. An evaluation of room-temp. tensile properties and high-temp. **creep strength** of IMI-829 near-.**alpha**. and Ti-25Al-10Nb-3B-1Mo Ti3Al alloys was performed. Prealloyed powders **produced** by the plasma rotating electrode process were compacted by rapid omnidirectional consolidation. Properly heat treated IMI-829 compacts performed as well as comparable near-.**alpha**. wrought alloys with similar microstructures. The as-compacted Ti3Al alloy showed substantially higher ductility in comparison with that of the wrought alloy. While the alloy demonstrated strength levels of .apprx.1100 MPa after heat treatment, the room temp. ductility was severely decreased to <1% under all heat treatment conditions. The Larsen-Miller comparison of the **creep strength** of the Ti3Al alloy placed it within the scatter **band** of values for the wrought alloy.

CC 56-12 (Nonferrous Metals and Alloys)

IT **69235-99-0**, IMI 829 128867-71-0
RL: PRP (Properties)
(mech. properties of powder-metallurgy)

IT **69235-99-0**, IMI 829
RL: PRP (Properties)
(mech. properties of powder-metallurgy)

L64 ANSWER 3 OF 5 HCAPLUS COPYRIGHT 2002 ACS

1991:564403 Document No. 115:164403 Effect of cooling rate on mechanical properties in **Zircaloy-4** alloy. Jeong, Yong Hwan; Choi, Chong Sool; Rheem, Karp Soon (Dep. Metall. Eng., Yonsei Univ., Seoul, 120-749, S. Korea). Taehan Kumsok Hakhoechi, 29(2), 104-11 (Korean) 1991. CODEN: TKHCDJ. ISSN: 0253-3847.

AB The effect of cooling rate on the mech. properties of **Zircaloy-4** alloy was studied for the specimens which were heated in the region of .beta.-phase and then cooled in various cooling media, such as ice brine, water, oil, air, and furnace atm. The ice brine and water quenching of the specimens resulted in higher **strength** and greater **strain** hardening rate than the oil quenching, air, and furnace cooling. The increase in the **strength** and **strain** hardening rate is attributed to the increase in stress required to move glide dislocations due to twins and tangled dislocations introduced during the quenching process, i.e., martensitic transformation. The **strength** and **strain** hardening rate were increased gradually as the cooling rate increased from furnace cooling (0.05.degree./s) to oil quenching (110.degree./s). The 2 properties are mainly controlled by .**alpha.-lath** size. From the microstructure and hardness, the ice brine and water quenched specimens resulted in faster recrystn. than the oil quenched and air cooled specimens. The ice brine quenched specimen was recrystd. through homogeneous nucleation, while the recrystn. of water-quenched specimen followed the bulge mechanism.

CC 56-12 (Nonferrous Metals and Alloys)

IT Martensitic structure
(**formation** of, in zirconium alloy, cooling rate effect on)

IT **11068-95-4, Zircaloy-4**
RL: PRP (Properties)
(mech. properties of, cooling rate effect on)

IT **11068-95-4, Zircaloy-4**
RL: PRP (Properties)
(mech. properties of, cooling rate effect on)

L64 ANSWER 4 OF 5 HCAPLUS COPYRIGHT 2002 ACS
1988:191111 Document No. 108:191111 Effect of substructure **formed** in prior .beta. grain on crack initiation and propagation **toughness** of Ti-6Al-2Sn-4Zr-6Mo alloy. Niinomi, Mitsuo; Inagaki, Ikuhiro; Kobayashi, Toshiro (Toyohashi Univ. Technol., Toyohashi, 440, Japan). Tetsu to Hagane, 74(3), 543-50 (Japanese) 1988. CODEN: TEHAA2. ISSN: 0371-6279.

AB The instrumented Charpy impact test, static, and dynamic fracture **toughness** tests were carried out on Ti-6Al-2Sn-4Zr-6Mo alloy in which the prior .beta. grain size was changed by heat treatments. The elongation, crack initiation, and propagation **toughness** increased with the slight decrease in strength in the specimens with the increased prior .beta. grain size and prolonged soln. treatment time in the .beta. region. The crack propagation **toughness** increased remarkably. The colony size, the width of grain boundary **.alpha** ., and the width and spacing of Widmanstaetten **.alpha**. also increased, but the subcolony spacing decreased with the increase in the prior .beta. grain size. The increase in the crack initiation **toughness** was mainly caused by the increase in the Widmanstaetten **.alpha. lath** or **lath** spacing. The increase in the crack propagation **toughness** was caused by the deflection of the crack pass, which was brought by the decrease in the intersubcolony spacing. The intersubcolony spacing decreased with the increase in the no. of **.alpha.** nucleation sites during diffusion-controlled . **alpha**. .fwdarw. .beta. transformation; such nucleation sites increased with the increase in the prior .beta. grain size. In such a situation, **.alpha.** nucleated in the interior of the .beta. grain and it increased its no. by the introduction of the working strain.

CC 56-12 (Nonferrous Metals and Alloys)

ST titanium alloy cracking **toughness** Widmanstaetten; aluminum titanium cracking **toughness** structure; tin titanium cracking **toughness** structure; zirconium titanium cracking **toughness** structure; molybdenum titanium cracking **toughness** structure

IT Widmanstaetten structure
(in titanium-aluminum-tin-zirconium-molybdenum alloy, crack initiation and propagation **toughness** in relation to)

IT **52293-96-6**, Ti6Al2Sn4Zr6Mo
RL: USES (Uses)
(crack initiation and propagation **toughness** of, substructure **formed** in .beta.-grains effect on)

IT **52293-96-6**, Ti6Al2Sn4Zr6Mo
RL: USES (Uses)
(crack initiation and propagation **toughness** of, substructure **formed** in .beta.-grains effect on)

L64 ANSWER 5 OF 5 HCAPLUS COPYRIGHT 2002 ACS
1976:466888 Document No. 85:66888 State of the surface layers of **manufactured** objects after polishing with diamond **strips** . Bekrenev, A. N.; Aleksentsev, E. I. (Kuibyshev, USSR). Sint. Almazy (6), 20-3 (Russian) 1975. CODEN: SIALBI.

AB The cutting zone temps. for OT4 [12633-22-6], VT9 [**12633-32-8**], and VT20 [**12670-26-7**] Ti alloys were 620-730, 540-690, and 400-580.degree. during polishing with corundum [1302-74-5], Si carbide, and diamond **strips**, resp. Corresponding **fatigue strengths** of polished VT9 alloy were 48-50, 51-53, and 53-55 kg/mm2. The strength was 42-45 kg/mm2 prior to polishing. The .beta.-. **alpha**. transformation occurred during polishing. The extent was the highest during polishing with diamond **strips** because of low surface work hardening. Tensile stress developed in Cr bronze

[12672-11-6] contg. 0.8 wt.% Cr during polishing with corundum **strips**, and compressive stress developed during polishing with diamond **strips**.
CC 56-7 (Nonferrous Metals and Alloys)
IT 12633-22-6 **12633-32-8 12670-26-7**
RL: PROC (Process)
(polishing of, with corundum and diamond and silicon carbide, surface state in relation to)
IT 409-21-2, uses and miscellaneous
RL: USES (Uses)
(polishing with, of titanium alloys, effect on **fatigue strength**)
IT **12633-32-8 12670-26-7**
RL: PROC (Process)
(polishing of, with corundum and diamond and silicon carbide, surface state in relation to)

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L67 ANSWER 1 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Influence of a zirconia layer on the mechanical behavior of **Zircaloy-4** cladding and thimble **tubes**

L67 ANSWER 2 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Tensile test of hydrided **zircaloy**

L67 ANSWER 3 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Evaluation of mechanical properties of hydrided cladding by using modified ring tensile test

L67 ANSWER 4 OF 42 HCAPLUS COPYRIGHT 2002 ACS
002 ACS
Cladding **tube** with zirconium liner for nuclear fuel rod and its
*manufacture***

L67 ANSWER 6 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Influence of cladding microstructure on the low enthalpy failures in RIA simulation tests

L67 ANSWER 7 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Application of blasting-ball treatment in lengthing services life of some nuclear facilities

L67 ANSWER 8 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Recent ABB BWR SVEA fuel failure experience

L67 ANSWER 9 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Recent ABB BWR failure experience

L67 ANSWER 10 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Zirconium alloy E635 as a material for fuel rod cladding and other components of VVER and RBMK cores

L67 ANSWER 11 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI BWR fuel secondary degradation status

L67 ANSWER 12 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Reactor fuel cladding **pipe** with zirconium-liner and its **manufacture**

L67 ANSWER 13 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Optimization of PWR behavior of stress-relieved **Zircaloy-4** cladding **tubes** by improving the **manufacturing** and inspection process

L67 ANSWER 14 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Development of new zirconium alloys for PWR fuel rod cladding

L67 ANSWER 15 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Development of new ferritic steels as cladding material for metallic fuel fast breeder reactor

L67 ANSWER 16 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI **Manufacture** of zirconium-alloy **pipes** for **cladding nuclear fuels**

L67 ANSWER 17 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI **Stress**-corrosion cracking **resistant** zirconium alloy **tubes** for **cladding** of **nuclear fuel**

L67 ANSWER 18 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI **Stress**-corrosion cracking **resistant** zirconium alloy **tubes** for **cladding** of **nuclear fuel**

L67 ANSWER 19 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI **Stress**-corrosion cracking **resistant** zirconium alloy **tubes** for **cladding** of **nuclear fuel**

L67 ANSWER 20 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Fatigue behavior of neutron irradiated **Zircaloy**-2 fuel cladding **tubes**

L67 ANSWER 21 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI PWC-11 fuel pin development for SP-100

L67 ANSWER 22 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Determination of plastic anisotropy of zirconium alloy cladding

L67 ANSWER 23 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI **Manufacture** of zirconium alloy **tubes**

L67 ANSWER 24 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Behavior of braze heat-affected zone in CANDU fuel sheaths

L67 ANSWER 25 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Effect of chemical composition on corrosion resistance of **Zircaloy** fuel cladding **tube** for BWR

L67 ANSWER 26 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Mechanical properties of zirconium alloy cladding **tubes** and critical fuel element power ramps

L67 ANSWER 27 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Heat-resistant steels

L67 ANSWER 28 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Nuclear fuel element

L67 ANSWER 29 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI **Zircaloy tube** for **nuclear fuel cladding**

L67 ANSWER 30 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Cladding **tube** for reactor fuel element

L67 ANSWER 31 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Evaluation of the resistance of irradiated zirconium-liner cladding to iodine-induced stress corrosion cracking

L67 ANSWER 32 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Fuel rods for nuclear reactors

L67 ANSWER 33 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Iodine-induced stress corrosion cracking of copper-barrier **Zircaloy-4 tubes**

L67 ANSWER 34 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Beta-quenching of **Zircaloy** cladding **tubes** in intermediate or final size - methods to improve corrosion and mechanical properties

L67 ANSWER 35 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Rupture criterion for **Zircaloy** cladding, when swelling in a reactor transient, calculated from free energy conditions

L67 ANSWER 36 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Is air a suitable environment for simulation of **Zircaloy** /steam-high temperature-oxidation within engineering experiments?

L67 ANSWER 37 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI The effects of texture and surface condition on the iodine stress corrosion cracking susceptibility of unirradiated **Zircaloy**-2

L67 ANSWER 38 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Behavior of unirradiated zirconium-lined and copper-plated **Zircaloy**-2 **tubing** under simulated PCI conditions

L67 ANSWER 39 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Design of an irradiation device for the determination of the in-pile creep behavior of **Zircaloy** cladding **tubes** under internal and external overpressure, in FRG-2

L67 ANSWER 40 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI Creep anisotropy of **Zircaloy** cladding **tubes**

L67 ANSWER 41 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI **Fabrication** technology and quality for **Zircaloy** fuel-cladding **tubes**

L67 ANSWER 42 OF 42 HCAPLUS COPYRIGHT 2002 ACS
TI **Fabrication** techniques and quality of **Zircaloy** cladding **tubes**

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L67 ANSWER 1 OF 42 HCAPLUS COPYRIGHT 2002 ACS
2002:392512 Document No. 137:12162 Influence of a zirconia layer on the mechanical behavior of **Zircaloy-4** cladding and thimble **tubes**. Berat-Robert, Laurence; Pelchat, Jacques; Limon, Roger; Maury, Roger; Pele, Jacques; Cappelaere, Chantal; Prioul, Claude; Bouffioux, Pol; Diz, Jesus (Commissariat a l'Energie Atomique/ Saclay, Gif sur Yvette Cedex, 91191, Fr.). Proceedings of the International Topical Meeting on LWR Fuel Performance, 10th, Park City, UT, United States, Apr. 10-13, 2000, 90-100. American Nuclear Society: La Grange Park, Ill. ISBN: 0-89448-656-X (English) 2000. CODEN: 69CPAO.

AB For std. PWR fuel assemblies, in reactor conditions, the microstructure of **Zircaloy-4** alloy is progressively modified by irradn. Cladding and thimble **tubes** also undergo corrosion: oxidn. and hydriding. This phenomenon induces a redn. of the safe metal thickness and may involve changes of the mech. properties of the cladding. This work aims at improving the knowledge and understanding of the zirconia effect in the mech. behavior of **Zircaloy-4** claddings under reactor and dry storage conditions. Up to now, irradn., hydriding and oxidizing effects have not often been studied sep. That is why the study only focuses on the oxidizing effects. To simplify testing conditions, unirradiated samples taken from recrystd. **Zircaloy-4** thimble and cladding **tubes** are used. They are previously oxidized under CO2 atmosphere. In order to det. the effect of the oxide behavior of the **Zircaloy-4 tubes** for different mech. loadings, several kinds of mech. tests are carried out: axial tensile tests, axial creep tests, burst tests and internal pressure creep tests. For these tests, samples with various oxide thicknesses are used (between 1 and 30 pm on each side). Major reinforcement due to the oxide layer occurs under axial loading. For example, for axial creep tests, the measured strain after 10 days for a **tube** with a 10 .mu.m oxide thickness is 4 times less than for the unoxidized specimen. The same reinforcement phenomenon is obsd. at the beginning of axial tensile tests. Tests under bi-axial loading do not reveal so much influence by the oxide layer. The circumferential plastic strain of an oxidized **tube** is higher than the strain of an unoxidized one. During burst tests, for example, an oxidized **tube** provides results comparable to those obtained with a **tube** whose thickness corresponds to the remaining unoxidized cladding thickness. After some mech. tests, a local characterization of the oxide layer is achieved by S.E.M. observations to elucidate zirconia behavior and to establish which mechanisms influence oxidized **tube** deformation. For samples tested under axial creep load, the oxide layers are cracked perpendicular to the applied load. The zirconia layer reduces the **tube** creep and the **Zircaloy** strain seems to be closely controlled by oxide cracking. In the case of samples tested under internal pressure creep load, the oxide layers are largely cracked parallel to the **tube** axis, that is perpendicular to the maximal principal stress. The oxidized **tubes** behave like thinner **tubes** because of the extensive embrittlement of the oxide layer. These different tests underline the fact that the influence of the oxide layer varies according to the type of applied load. On the one hand, in the case of axial loading, zirconia has a beneficial influence on cladding mech. **resistance** and reduces **creep** mechanism. However, zirconia does not demonstrate such a favorable effect under bi-axial loading.

CC 71-5 (Nuclear Technology)
Section cross-reference(s): 56

ST zirconia **Zircaloy** cladding thimble **tube** PWR fuel assembly

IT Fuel assemblies
(PWR; influence of zirconia layer on mech. behavior of **Zircaloy -4** cladding and thimble **tubes** in)
IT Plastic deformation
(circumferential plastic strain of oxidized **tube** is higher than strain of unoxidized one.)
IT Pressurized water nuclear reactors
(fuel assemblies; influence of zirconia layer on mech. behavior of **Zircaloy-4** cladding and thimble **tubes** in)
IT Mechanical properties
(influence of zirconia layer on mech. behavior of **Zircaloy-4** cladding and thimble **tubes**)
IT **Nuclear fuel** element **cladding**
(influence of zirconia layer on mech. behavior of **Zircaloy-4** cladding and thimble **tubes**.)
IT Microstructure
(microstructure of **Zircaloy-4** alloy is progressively modified by irradn.)
IT Embrittlement
(oxidized **tubes** behave like thinner **tubes** because of extensive embrittlement of oxide layer.)
IT 1314-23-4, Zirconia, **formation** (nonpreparative)
RL: FMU (Formation, unclassified); FORM (Formation, nonpreparative)
(influence of zirconia layer on mech. behavior of **Zircaloy-4** cladding and thimble **tubes**)
IT **11068-95-4, Zircaloy-4**
RL: PRP (Properties)
(influence of zirconia layer on mech. behavior of **Zircaloy-4** cladding and thimble **tubes**)
IT **11068-95-4, Zircaloy-4**
RL: PRP (Properties)
(influence of zirconia layer on mech. behavior of **Zircaloy-4** cladding and thimble **tubes**)

L67 ANSWER 2 OF 42 HCAPLUS COPYRIGHT 2002 ACS
2002:94997 Document No. 136:174308 Tensile test of hydrided **zircaloy**. Kuroda, Masatoshi; Yamanaka, Shinsuke; Setoyama, Daigo; Uno, Masayoshi; Takeda, Kiyoko; Anada, Hiroyuki; Nagase, Fumihisa; Uetsuka, Hiroshi (Department of Nuclear Engineering, Graduate School of Engineering, Osaka University, Suita, 565-0871, Japan). Journal of Alloys and Compounds, 330-332, 404-407 (English) 2002. CODEN: JALCEU. ISSN: 0925-8388. Publisher: Elsevier Science S.A..
AB To examine the influence of pptd. Zr hydride on the failure behavior and **fracture strength** of light H2O reactor (LWR) cladding **tubes**, tensile tests were performed at room temp. for nonhydrided and hydrided **Zircaloy** sheet-type specimens with gauge section of 10.0.times.5.0 mm and thicknesses of 0.5, 1.0, 1.5, 2.0, 2.5, and 3.0 mm. For specimens with thickness >2.5 mm, the ultimate tensile strength of the specimens appeared to be independent of thickness, which implied that plane strain condition was attained. For the specimen with 2.5 mm thickness, the ultimate tensile strength increased slightly with increasing av. H concn. Through microscopic observation of the hydrided specimen surface by SEM, matrix/hydride de-bonding was not **generated** but micro-cracks perpendicular to the axial direction were **produced** at the hydride layer.
CC 71-3 (Nuclear Technology)
Section cross-reference(s): 56
ST tensile hydrided **zircaloy** LWR cladding
IT Light-water nuclear reactors
Nuclear fuel element **cladding**

(failure behavior and **fracture strength** of light
H2O reactor cladding)

IT Fracture (materials)
(influence of pptd. Zr hydride on failure behavior and **fracture strength**)

IT Tensile strength
(tensile test of hydrided **zircaloy**)

IT **11068-95-4**
RL: PRP (Properties)
(Tensile test of hydrided **zircaloy**)

IT 11105-16-1, Zirconium hydride
RL: PRP (Properties)
(influence of pptd. Zr hydride on failure behavior and **fracture strength**)

IT 12184-88-2, Hydride
RL: PEP (Physical, engineering or chemical process); PROC (Process)
(tensile test of hydrided **zircaloy**)

IT **11068-95-4**
RL: PRP (Properties)
(Tensile test of hydrided **zircaloy**)

L67 ANSWER 3 OF 42 HCAPLUS COPYRIGHT 2002 ACS

2002:20069 Document No. 136:223054 Evaluation of mechanical properties of hydrided cladding by using modified ring tensile test. Kitano, Koji; Fuketa, Toyoshi; Uetsuka, Hiroshi (Dep. Reactor Safety Res., Nuclear Safety Res. Center, Tokai Res. Establishment, Japan Atomic Energy Research Institute, Tokai-mura, Naka-gun, Ibaraki-ken, Japan). JAERI-Research, 2001-041, i-iv, 1-24 (Japanese) 2001. CODEN: JERIE4.

AB Results from the pulse irradn. tests at NSRR indicated that failure of high burn-up fuel rod under Reactivity Initiated Accident conditions occurs due to pellet/cladding mech. interaction (PCMI). The authors performed modified ring tensile test on cladding **tube** samples with artificially **made** hydride rim to evaluate the influence of hydride rim on mech. properties of cladding in hoop direction. Fracture strain reduces with hydride rim thickness at room temp. because cracks could **generate** in brittle hydride rim region at the beginning stage of deformation. At elevated temp. (573 K), fracture strain varied depending not only on thickness of hydride rim but also on hydride d. in rim region. The specimen with hydride rim of low hydride d. showed larger fracture strain regardless of the hydride rim thickness. This may be attributed to ductile-brittle transition of hydride rim region with temp. increase. The rim of low hydride d. could be not brittle but ductile at 573 K. Thus probably fracture strain of the specimen with thick hydride rim becomes larger when the hydride d. in rim region is low.

CC 71-5 (Nuclear Technology)
Section cross-reference(s): 56

ST hydride **nuclear** reactor **fuel cladding**
tensile strain

IT Fracture **toughness**
Nuclear fuel element **cladding**
Nuclear reactor accident
Strain
Tensile strength
(evaluation of mech. properties of hydrided cladding by using modified ring tensile test)

IT **11068-95-4, Zircaloy 4**
RL: PRP (Properties)
(evaluation of mech. properties of hydrided cladding by using modified ring tensile test)

IT **11068-95-4, Zircaloy 4**

RL: PRP (Properties)
(evaluation of mech. properties of hydrided cladding by using modified ring tensile test)

L67 ANSWER 4 OF 42 HCAPLUS COPYRIGHT 2002 ACS
2000:808616 Document No. 133:341714 Cladding for use in nuclear reactors having improved resistance to cracking and corrosion. Admson, Ronald Bert; Lutz, Daniel Reese; Marlowe, Mickey Orville; Schardt, John Frederick; Williams, Cedric David (General Electric Company, USA). Eur. Pat. Appl. EP 1052650 A1 20001115, 16 pp. DESIGNATED STATES: R: AT, BE, CH, DE, DK, ES, FR, GB, GR, IT, LI, LU, NL, SE, MC, PT, IE, SI, LT, LV, FI, RO. (English). CODEN: EPXXDW. APPLICATION: EP 2000-304016 20000512. PRIORITY: US 1999-312021 19990514.

AB An improved fuel element for use in a nuclear reactor comprised of a central core of nuclear material, which is surrounded by a composite cladding. The cladding has an outer metallic **tubular** portion comprised of well-known cladding alloys used for such purposes. Metallurgically bonded to the outer metallic **tubular** portion is a com. pure zirconium microalloyed with a controlled quantity of iron. The zirconium microalloyed with iron **produce** an inner metallic barrier having a beneficial balance between **stress** corrosion crack **resistance** and corrosion resistance while retaining other beneficial properties of pure zirconium, such as ductility.

IC ICM G21C003-07
ICS G21C003-20

CC 71-5 (Nuclear Technology)

ST **cladding nuclear fuel** resistance cracking corrosion zirconium alloy iron

IT **Nuclear fuels**
Nuclear reactors
Stress corrosion cracking
(**cladding** for use in nuclear reactors having improved resistance to cracking and corrosion)

IT **11068-94-3, Zircaloy** 2
RL: NUU (Other use, unclassified); USES (Uses)
(use in nuclear reactor fuel elements)

IT **11068-94-3, Zircaloy** 2
RL: NUU (Other use, unclassified); USES (Uses)
(use in nuclear reactor fuel elements)

L67 ANSWER 5 OF 42 HCAPLUS COPYRIGHT 2002 ACS
2000:585568 Document No. 133:184582 Cladding **tube** with zirconium liner for nuclear fuel rod and its **manufacture**. Nakatsukasa, Masafumi (Japan Nuclear Fuel Development Co., Ltd., Japan; Toshiba Corp.; Hitachi, Ltd.). Jpn. Kokai Tokkyo Koho JP 2000230993 A2 20000822, 6 pp. (Japanese). CODEN: JKXXAF. APPLICATION: JP 1999-30795 19990209.

AB The cladding **tube** comprises a **Zr alloy tube** with a high-purity Zr liner contg. 400-1000 ppm (in total) Fe and/or Cr and .ltoreq.400 ppm O as additives and .ltoreq.1000 ppm inevitable impurities. Av. crystal grain size of the Zr liner is 10-13 .mu.m, and at least part of the inside of the liner has oxidized layer. In **manufg**. the cladding **tube** by inserting a Zr **pipe** into a **Zr alloy tube**, bonding by metallurgy, and alternately repeating cold rolling and annealing, oxidn. of the cladding **tube** is carried out in finish annealing process. The cladding **tube** has high **resistance** to **stress** corrosion cracking, and is useful for light-water nuclear reactors (LWR).

IC ICM G21C003-20
ICS G21C003-06

CC 71-4 (Nuclear Technology)
Section cross-reference(s): 56
ST zirconium liner **cladding tube nuclear fuel** rod; oxidn annealing zirconium **cladding tube nuclear fuel**; **stress** corrosion cracking **resistance cladding tube nuclear fuel**
IT **Nuclear fuels**
(**Zr alloy cladding tube** with **Zr** liner contg. Fe and/or Cr and O for high **resistance** to **stress** corrosion cracking for nuclear fuel rod)
IT Oxidation
(annealing and oxidn. in **manuf**. of **Zr alloy** cladding **tube** with **Zr** liner for nuclear fuel rod)
IT Coating materials
(anticorrosive; **Zr alloy** cladding **tube** with **Zr** liner contg. Fe and/or Cr and O for high **resistance** to **stress** corrosion cracking for nuclear fuel rod)
IT Coating materials
(linings; **Zr alloy** cladding **tube** with **Zr** liner contg. Fe and/or Cr and O for high **resistance** to **stress** corrosion cracking for nuclear fuel rod)
IT Containers
(**tubes**; **Zr alloy** cladding **tube** with **Zr** liner contg. Fe and/or Cr and O for high **resistance** to **stress** corrosion cracking for nuclear fuel rod)
IT 7440-67-7, Zirconium, uses **11068-94-3 12614-57-2 141825-38-9**
RL: DEV (Device component use); PEP (Physical, engineering or chemical process); PRP (Properties); PROC (Process); USES (Uses)
(**Zr alloy** cladding **tube** with **Zr** liner contg. Fe and/or Cr and O for high **resistance** to **stress** corrosion cracking for nuclear fuel rod)
IT 7782-44-7, Oxygen, uses
RL: MOA (Modifier or additive use); USES (Uses)
(microalloying element; **Zr alloy** cladding **tube** with **Zr** liner contg. Fe and/or Cr and O for high **resistance** to **stress** corrosion cracking for nuclear fuel rod)
IT **11068-94-3 12614-57-2 141825-38-9**
RL: DEV (Device component use); PEP (Physical, engineering or chemical process); PRP (Properties); PROC (Process); USES (Uses)
(**Zr alloy** cladding **tube** with **Zr** liner contg. Fe and/or Cr and O for high **resistance** to **stress** corrosion cracking for nuclear fuel rod)

L67 ANSWER 6 OF 42 HCAPLUS COPYRIGHT 2002 ACS
2000:242474 Document No. 133:10033 Influence of cladding microstructure on the low enthalpy failures in RIA simulation tests. Garde, A. M. (ABB Combustion Engineering Nuclear Fuel, Windsor, CT, 06095, USA). ASTM Special Technical Publication, STP 1354(Zirconium in the Nuclear Industry: Twelfth International Symposium, 1998), 234-255 (English) 2000. CODEN: ASTTA8. ISSN: 0066-0558. Publisher: ASTM.
AB A review with 30 refs. Welding used during the **prepn**. of specimens for Reactivity Initiated Accident (RIA) simulation testing from fuel rods previously irradiated in reactors introduces the following 4 microstructural changes in the specimen: (a) annealing of irradn. damage, (b) a change from an .alpha.-phase structure of **Zircaloy** to a

transformed .beta. structure in the cladding, (c) dissoln. of the hydride rim **formed** under the oxide on the cladding **tube** outer surface during normal irradn. and possible radial-oriented hydride repptn. at the transformed .beta. platelet boundaries, and (d) repptn. of 2nd-phase particles previously dissolved due to radiation damage. A 5th factor, the change in the texture of **Zircaloy**, is also introduced due to the welding operation. The possible effect of these 5 changes on the specimen fracture **toughness** and failure enthalpy is evaluated. The published data on the mech. properties of irradiated and unirradiated transformed beta structure of **Zircaloy** charged with H are reviewed to evaluate the impact of the 5 anticipated microstructural changes on the failure enthalpy. The available RIA simulation test fracture data with low-failure enthalpy are reviewed. The limited failure path information available appears to indicate that microstructural factors have contributed to the low-enthalpy failures. The applicability of the results from the low-enthalpy RIA test failures to Light Water Reactor (LWR) nuclear fuel should be based on the representativeness of the RIA specimen microstructure to that of the LWR fuel cladding.

CC 71-0 (Nuclear Technology)
Section cross-reference(s): 56

IT Fracture **toughness**
Light-water nuclear reactors
Nuclear fuel element **cladding**
Nuclear reactor accident
Simulation and Modeling, physicochemical
Structural phase transition
Welding of metals
(influence of cladding microstructure on low enthalpy failures in RIA simulation tests)

IT **11068-94-3, Zircaloy**-2 **11068-95-4, Zircaloy-4**
RL: PRP (Properties)
(influence of cladding microstructure on low enthalpy failures in RIA simulation tests)

IT **11068-94-3, Zircaloy**-2 **11068-95-4, Zircaloy-4**
RL: PRP (Properties)
(influence of cladding microstructure on low enthalpy failures in RIA simulation tests)

L67 ANSWER 10 OF 42 HCAPLUS COPYRIGHT 2002 ACS
1997:71449 Document No. 126:230459 Zirconium alloy E635 as a material for fuel rod cladding and other components of VVER and RBMK cores. Nikulina, Antonina V.; Markelov, Vladimir A.; Peregud, Mikhail M.; Bibilashvili, Yury K.; Kotrekhov, Vladimir A.; Lositsky, Anatoly F.; Kuzmenko, Nikolay V.; Shevnin, Yuriy P.; Shamardin, Valentin K.; Kobylyansky, Gennady P.; Novoselov, Andrey E. (A. A. Bochvar All-Russia Scientific Res. Inst. Inorganic Maters., Moscow, 123060, Russia). ASTM Special Technical Publication, STP 1295(Zirconium in the Nuclear Industry: Eleventh International Symposium, 1995), 785-804 (English) 1996. CODEN: ASTTA8. ISSN: 0066-0558. Publisher: American Society for Testing and Materials.

AB Data are given on **Zr alloy** E635 (**Zr** -1.2Sn-1Nb-0.4Fe), developed in Russia as a fuel rod cladding and other component material for use in cores of WWR and RBMK types. The alloy is much superior to binary alloys with 1.0 and 2.5% Nb and **Zircaloys** in terms of its **resistance** to irradn.-induced **creep** and growth and nodular corrosion. The creep rate of the alloy is slightly dependent on irradn. temp., stress, n fluence, and n d. The alloy is subject to substantial irradn. hardening while retaining its high-percent

elongation. Corrosion, **creep**, and growth **resistances**
are slightly dependent on the structure of components (alloy, final
product). Based on the previously studied influence of
impurities, structure, heat treatment, and working schedules, the technol.
processes were designed and mastered com. for **fabrication** of
tubes, bars, strips, and fuel rod claddings from this alloy.
Components are **produced** com. Fuel assemblies with fuel rods
clad in the E635 alloy were successfully tested in the RBMK reactor at the
Leningrad NPP as well as in exptl. reactors under WWR-1000 conditions.
Today, the E635 alloy is recommended as a promising material for use in
cores of WWR-1000 and WWR of new **generations** as well as
RBMK-type reactors having a longer fuel cycle.

CC 71-5 (Nuclear Technology)
Section cross-reference(s): 56

IT **Nuclear fuel** element **cladding**
(zirconium alloy E635 as material for fuel rod cladding and other
components of VVER and RBMK cores)

IT **150287-03-9**, E635
RL: NUU (Other use, unclassified); PRP (Properties); USES (Uses)
(zirconium alloy E635 as material for fuel rod cladding and other
components of VVER and RBMK cores)

IT **150287-03-9**, E635
RL: NUU (Other use, unclassified); PRP (Properties); USES (Uses)
(zirconium alloy E635 as material for fuel rod cladding and other
components of VVER and RBMK cores)

L67 ANSWER 12 OF 42 HCAPLUS COPYRIGHT 2002 ACS
1995:982773 Document No. 124:39886 Reactor fuel cladding **pipe** with
zirconium-liner and its **manufacture**. Nakatsuka, Masafumi
(Nippon Kakunenryo Kaihatsu Kk, Japan). Jpn. Kokai Tokkyo Koho JP
07248391 A2 19950926 Heisei, 4 pp. (Japanese). CODEN: JKXXAF.
APPLICATION: JP 1994-42388 19940314.

AB The cladding **pipe** has an inside structure consisting of a
Zr alloy lining and a layer preventing H-permeation.
The **pipe** may be coated with the layer preventing H-permeation.
The layer may be black monoclinic Zr oxide. In the **manuf.**, the
layer is obtained by oxidn. the **pipe** in a vapor at atm.
pressure. The **pipe** shows good **resistance** to H-
embrittlement.

IC ICM G21C003-20
ICS G21C003-06

CC 71-5 (Nuclear Technology)
Section cross-reference(s): 56

ST reactor fuel **pipe** zirconium lining; hydrogen
embrittlement resistance zirconium **pipe**

IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, reactor fuel cladding **pipe** with
Zr-liner and Zr oxide and its **manuf.**)

IT 1314-23-4P, Zirconium oxide, uses
RL: PNU (Preparation, unclassified); TEM (Technical or engineered material
use); PREP (Preparation); USES (Uses)
(reactor fuel cladding **pipe** with Zr-liner and Zr oxide and
its **manuf.**)

IT 7440-67-7, Zirconium, uses **11068-94-3, Zircaloy**-2
RL: TEM (Technical or engineered material use); USES (Uses)
(reactor fuel cladding **pipe** with Zr-liner and Zr oxide and
its **manuf.**)

IT **11068-94-3, Zircaloy**-2
RL: TEM (Technical or engineered material use); USES (Uses)
(reactor fuel cladding **pipe** with Zr-liner and Zr oxide and

its **manuf**.)

L67 ANSWER 13 OF 42 HCAPLUS COPYRIGHT 2002 ACS
1995:700115 Document No. 123:239819 Optimization of PWR behavior of stress-relieved **Zircaloy-4** cladding **tubes** by improving the **manufacturing** and inspection process. Mardon, Jean-Paul; Charquet, Daniel; Senevat, Jean (FRAMATOME Nuclear Fuel Division, FRAGEMA, Lyon, 69006, Fr.). ASTM Spec. Tech. Publ., STP 1245(Zirconium in the Nuclear Industry: Tenth International Symposium, 1993), 328-48 (English) 1994. CODEN: ASTTA8. ISSN: 0066-0558.

AB With the aim of optimizing the basic properties of stress-relieved **Zircaloy-4** cladding **tubes**, particularly those that **make** it possible to push back the initial technol. limits that may be encountered, and of reducing the scatter of those properties and enhancing **tube** quality, the role of the main parameters involved in **manufg**. the ingot, Trex, and cladding **tube** was evaluated on an industrial scale. Large-sized **tube** lots were **produced** under controlled **manufg**. conditions, then characterized by out-of-pile test results (short-and long-term corrosion, stress corrosion cracking (SCC), creep, mech., and structural properties) on finished **tubes**. For the studied parameters (chem. compn., no. of melt, quench rate, accumulated annealing parameter, the .SIGMA.A factor, surface condition (outside and inside diams.), and finished **tube** quality), this role is indeed important but complex due to the highly interactive nature of the variables studied. Adjustment of the chem. compn. within ASTM limits enables generalized corrosion resistance to be enhanced and irradn. growth to be minimized. A significant decrease of the obsd. scatter in corrosion and mech. properties is obtained by optimization of the .SIGMA.A range, the quenching rate, and the final heat treatment. The optimum seems to be reached for a final treatment at the highest possible temp. compatible with the stress-relieved state, corresponding to an av. ppt. size and .SIGMA.A. Also, by adding anneals upstream in the process, a further increase in this .SIGMA.A no longer seems to have a significant effect on generalized corrosion. Finally, extensive efforts were employed in the pilgering, surface **prepn**. (outside diam. polishing, flush-pickling), and examn. method (UT, EC) leading to a sizable improvement in SCC resistance and to a redn. in scatter for finished **tubes**. The result of these optimizations was implemented in the current AFA-2G, that shows that under irradn. a 30% corrosion gain is reached after 3 cycles, without degrading **creep strength** or growth. The intrinsic effect of Sn on generalized corrosion resistance under irradn. was also confirmed on this occasion.

CC 71-5 (Nuclear Technology)
Section cross-reference(s): 56

ST optimization PWR behavior stress relieved **Zircaloy**; cladding **tube** improving **Zircaloy** optimization

IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, Optimization of PWR behavior of stress-relieved **Zircaloy-4** cladding **tubes** by improving the **manufg**. and inspection process)

IT 7440-31-5, Tin, processes **11068-95-4**, **Zircaloy-4** 12586-31-1, Neutron
RL: PEP (Physical, engineering or chemical process); PROC (Process)
(Optimization of PWR behavior of stress-relieved **Zircaloy-4** cladding **tubes** by improving the **manufg**. and inspection process)

IT **11068-95-4**, **Zircaloy-4**
RL: PEP (Physical, engineering or chemical process); PROC (Process)
(Optimization of PWR behavior of stress-relieved **Zircaloy-**

4 cladding **tubes** by improving the **manufg.**
and inspection process)

L67 ANSWER 14 OF 42 HCAPLUS COPYRIGHT 2002 ACS
1994:665983 Document No. 121:265983 Development of new zirconium alloys for PWR fuel rod cladding. Mardon, Jean Paul; Charquet, Daniel; Senevat, Jean (Framatome Nuclear Fuel, Lyon, 69456, Fr.). Proc. Int. Top. Meet. Light Water React. Fuel Perform., 643-9. Am. Nucl. Soc.: La Grange Park, Ill. (English) 1994. CODEN: 60DGA9.

AB Within the scope of research into PWR fuel rod cladding materials with higher performance than **Zircaloy-4**, four zirconium alloys were **produced** and transformed into solid-wall and coextruded **tubes** according to industrial process outlines. These alloys break down into two groups : three ultra low tin alloys contg. (Fe, Cr, Nb, O or V) and one ternary niobium - oxygen alloy. These solns. were characterized out of pile for corrosion, creep, SCC, mech. and structural properties, high-temp. creep, and in pile for corrosion, growth and creep. In relation to stress-relieved AFA-2G **Zircaloy-4** (low tin optimized **Zircaloy-4**), the new **Zr alloys** have exhibited better waterside corrosion resistance through three irradn. cycles (40 GWd/t). The results are not the same as those obtained in autoclave testing. The corrosion resistance found in autoclave testing is relatively smaller in the case of the ultra low tin alloys and higher for the ternary niobium - oxygen alloy. The monotonic influence from 0.5 % to 1.7 % tin on corrosion obsd. in autoclave is not seen under irradn., it is more marked between 1.7 and 1.2 % than between 1.2 and 0.5 %. The corrosion behavior of the zirconium alloys outside **Zircaloy-4** cannot be detd. solely on the basis of autoclave tests in water or steam; final judgement has to be confirmed by irradn. with relatively high burnup data. The other properties important for clad behavior like creep and growth are at least identical to those of AFA-2G **Zircaloy-4** and significant improvements for these two properties are obsd. for some of these alloys. For the coextruded **tubes**, corrosion resistance is dictated by the external layer alloy; the substrate can provide SCC resistance and the substrate/external layer assembly provides mech. **strength** and **creep** performance.

CC 71-5 (Nuclear Technology)

ST zirconium alloy PWR fuel rod cladding; **nuclear** reactor **fuel cladding**

IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, zirconium alloys for PWR)

IT **11068-95-4P, Zircaloy-4**
RL: PNU (Preparation, unclassified); PREP (Preparation)
(PWR fuel rod cladding)

IT **81029-19-8P 158634-69-6P,** Iron 0.2 niobium 0.5 oxygen 0.1 tin 0.5 zirconium 99 **158634-70-9P,** Chromium 0.1 iron 0.2 oxygen 0.2 tin 0.5 zirconium 99 **158634-71-0P,** Iron 0.6 oxygen 0.1 tin 0.5 vanadium 0.3 zirconium 99
RL: PNU (Preparation, unclassified); PRP (Properties); SPN (Synthetic preparation); PREP (Preparation)
(PWR fuel rod cladding)

IT **11068-95-4P, Zircaloy-4**
RL: PNU (Preparation, unclassified); PREP (Preparation)
(PWR fuel rod cladding)

IT **81029-19-8P 158634-69-6P,** Iron 0.2 niobium 0.5 oxygen 0.1 tin 0.5 zirconium 99 **158634-70-9P,** Chromium 0.1 iron 0.2 oxygen 0.2 tin 0.5 zirconium 99 **158634-71-0P,** Iron 0.6 oxygen 0.1 tin 0.5 vanadium 0.3 zirconium 99
RL: PNU (Preparation, unclassified); PRP (Properties); SPN (Synthetic

preparation); PREP (Preparation)
(PWR fuel rod cladding)

L67 ANSWER 16 OF 42 HCAPLUS COPYRIGHT 2002 ACS
1993:134573 Document No. 118:134573 **Manufacture** of zirconium-alloy **pipes** for **cladding nuclear fuels.**
Kikukawa, Tomokazu; Suda, Yoshitaka; Isobe, Takeshi (Mitsubishi Materials Corp., Japan). Jpn. Kokai Tokkyo Koho JP 04154943 A2 19920527 Heisei, 4 pp. (Japanese). CODEN: JKXXAF. APPLICATION: JP 1990-276862 19901016.

AB In the process, in which a **Zr-alloy pipe** is repeatedly extruded and annealed for recrystn., and is finally annealed for warp removal, the **pipe** is extruded while stress within the elastic limit of the alloy is applied in the axial direction of the **pipe**. The **Zr-alloy pipe** has increased **resistance** to **stress**-corrosion cracking.

IC ICM C22F001-18
ICS B21B021-00; G21C003-06

CC 71-5 (Nuclear Technology)

ST zirconium alloy **pipe cladding nuclear fuel**

IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, zirconium-alloy **pipes**, **manuf**. of)

IT Zirconium alloy, base
RL: PROC (Process)
(**pipes** from, for **cladding nuclear fuels**, **manuf**. of)

IT **89342-04-1**
RL: PROC (Process)
(**pipes** from, for **cladding nuclear fuels**, **manuf**. of)

IT **89342-04-1**
RL: PROC (Process)
(**pipes** from, for **cladding nuclear fuels**, **manuf**. of)

L67 ANSWER 17 OF 42 HCAPLUS COPYRIGHT 2002 ACS
1992:536009 Document No. 117:136009 **Stress**-corrosion cracking **resistant** zirconium alloy **tubes** for **cladding** of **nuclear fuel**. Mae, Yoshiharu; Isobe, Takeshi (Mitsubishi Materials K. K., Japan). Jpn. Kokai Tokkyo Koho JP 04099256 A2 19920331 Heisei, 6 pp. (Japanese). CODEN: JKXXAF. APPLICATION: JP 1990-212642 19900810.

AB The title **tubes** are **manufd**. by extruding **Zr alloy** to give raw **tubes**, Pilger rolling and recrystn. annealing of the raw **tubes** for .gtoreq. 1 times, resp., then final Pilger rolling and stress-relief annealing for .gtoreq.1 times to give the o.d. redn. rate 1-15%. Thus, an extruded **tube** from **Zr alloy** contg. Sn 1.5, Fe 0.2, and Cr 0.1% after the heat treatment showed good machinability and high **stress** -corrosion cracking **resistance**.

IC ICM C22F001-18
ICS B21B021-00; B21C001-22; G21C003-06

CC 56-11 (Nonferrous Metals and Alloys)
Section cross-reference(s): 71

ST zirconium alloy **tube** cracking resistance; **stress** corrosion cracking **resistance tube**; **nuclear fuel cladding** zirconium alloy **tube**

IT **Pipes** and **Tubes**
(zirconium alloy, **stress**-corrosion cracking-**resistant**

, **manuf**. of)
IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, zirconium alloy **tubes**, good
stress-corrosion cracking-**resistant**, **manuf**.
of)
IT Zirconium alloy, base
RL: PEP (Physical, engineering or chemical process); PROC (Process)
(**tubes**, **stress**-corrosion cracking **resistant**
, **manuf**. of, for **cladding** of **nuclear**
fuel)
IT **89342-04-1P**
RL: PEP (Physical, engineering or chemical process); PREP (Preparation);
PROC (Process)
(**tubes**, **stress**-corrosion cracking **resistant**
, **manuf**. of, for **cladding** of **nuclear**
fuel)
IT **89342-04-1P**
RL: PEP (Physical, engineering or chemical process); PREP (Preparation);
PROC (Process)
(**tubes**, **stress**-corrosion cracking **resistant**
, **manuf**. of, for **cladding** of **nuclear**
fuel)

L67 ANSWER 18 OF 42 HCAPLUS COPYRIGHT 2002 ACS
1992:536008 Document No. 117:136008 **Stress**-corrosion cracking
resistant zirconium alloy **tubes** for **cladding**
of **nuclear fuel**. Mae, Yoshiharu; Isobe, Takeshi
(Mitsubishi Materials K. K., Japan). Jpn. Kokai Tokkyo Koho JP 04099255
A2 19920331 Heisei, 4 pp. (Japanese). CODEN: JKXXAF. APPLICATION: JP
1990-212641 19900810.
AB The title **tubes** are **manufd.** by extruding of **Zr**
alloy to give raw **tubes**, Pilger rolling of the raw
tubes, and recrystn. annealing at 530-760.degree. under vacuum
atm. for .gtoreq. 1 times, resp., then final Pilger rolling and
stress-relief annealing at 430-490.degree. to give the o.d. redn. rate
1-15 %. Thus, an extruded **tube** from **Zr alloy**
contg. Sn 1.5, Fe 0.2, and Cr 0.1% after the heat treatment showed high
stress-corrosion cracking **resistance**.
IC ICM C22F001-18
ICS B21B021-00; B21C001-22; G21C003-06
CC 56-11 (Nonferrous Metals and Alloys)
Section cross-reference(s): 71
ST zirconium alloy **tube** cracking resistance; **stress**
corrosion cracking **resistance tube**; **nuclear**
fuel cladding zirconium alloy **tube**
IT **Pipes** and **Tubes**
(zirconium alloy for, **stress**-corrosion cracking-
resistant, **manuf**. of)
IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, zirconium alloys, **stress**-corrosion
cracking-**resistant**, **manuf**. of)
IT Zirconium alloy, base
RL: PEP (Physical, engineering or chemical process); PROC (Process)
(**tubes**, with good **stress**-corrosion cracking
resistance, **manuf**. of, for **cladding** of
nuclear fuel)
IT **89342-04-1P**
RL: PEP (Physical, engineering or chemical process); PREP (Preparation);
PROC (Process)
(**tubes**, **stress**-corrosion cracking **resistant**

, **manuf**. of, for **cladding** of **nuclear**
fuel)

IT **89342-04-1P**
RL: PEP (Physical, engineering or chemical process); PREP (Preparation); PROC (Process)
(**tubes**, **stress**-corrosion cracking **resistant**
, **manuf**. of, for **cladding** of **nuclear**
fuel)

L67 ANSWER 19 OF 42 HCAPLUS COPYRIGHT 2002 ACS
1992:536007 Document No. 117:136007 **Stress**-corrosion cracking **resistant** zirconium alloy **tubes** for **cladding** of **nuclear fuel**. Mae, Yoshiharu; Isobe, Takeshi (Mitsubishi Materials K. K., Japan). Jpn. Kokai Tokkyo Koho JP 04099254 A2 19920331 Heisei, 5 pp. (Japanese). CODEN: JKXXAF. APPLICATION: JP 1990-212640 19900810.

AB The title **tubes** are **manufd**. by extruding **Zr alloy** to give raw **tubes**, recrystn. annealing of the raw **tubes** under vacuum for .gtoreq. 1 times, , Pilger rolling for .gtoreq.1 times, and stress-relief annealing at 430-490.degree. to give the o.d. redn. rate 1-30 %. Thus, an extruded **tube** of **Zr alloy** contg. Sn 1.5, Fe 0.2, and Cr 0.1% showed high **stress**-corrosion cracking **resistance**.

IC ICM C22F001-18
ICS B21B021-00; B21C001-22; C22C016-00; G21C003-06

CC 56-11 (Nonferrous Metals and Alloys)
Section cross-reference(s): 71

ST zirconium alloy **tube** cracking resistance; **stress** corrosion cracking **resistance tube**; **nuclear fuel cladding** zirconium alloy **tube**

IT **Pipes** and **Tubes**
(zirconium alloy, with good **stress**-corrosion cracking
resistance, **manuf**. of)

IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, zirconium alloy **tubes**, **stress**
-corrosion cracking-**resistant**, **manuf**. of)

IT Zirconium alloy, base
RL: PEP (Physical, engineering or chemical process); PROC (Process)
(**tubes**, with good **stress**-corrosion cracking
resistance, **manuf**. of, for **cladding** of
nuclear fuel)

IT **89342-04-1P**
RL: PEP (Physical, engineering or chemical process); PREP (Preparation); PROC (Process)
(**tubes**, with good **stress**-corrosion cracking
resistance, **manuf**. of, for **cladding** of
nuclear fuel)

IT **89342-04-1P**
RL: PEP (Physical, engineering or chemical process); PREP (Preparation); PROC (Process)
(**tubes**, with good **stress**-corrosion cracking
resistance, **manuf**. of, for **cladding** of
nuclear fuel)

L67 ANSWER 20 OF 42 HCAPLUS COPYRIGHT 2002 ACS
1992:138218 Document No. 116:138218 Fatigue behavior of neutron irradiated **Zircaloy**-2 fuel cladding **tubes**. Nakatsuka, Masafumi; Kubo, Toshio; Hayashi, Yo (Nippon Nucl. Fuel Dev. Co., Ltd., Oarai, Japan). ASTM Spec. Tech. Publ., 1132(Zirconium Nucl. Ind.), 230-45 (English) 1991. CODEN: ASTTA8. ISSN: 0066-0558.

AB The effects of n irradn. and I as a corrosive fission **product** on the fatigue behavior of **Zircaloy**-2 fuel cladding **tubes** were investigated using 2 different types of test specimens to evaluate the **fatigue strength** of BWR fuel subjected to such variable loading conditions as load following or automatic frequency control operations. Fatigue life had a tendency to drop with increasing I partial pressure, reaching a satn. value .apprx.1/10 of that in an inert gas atm. Min. I partial pressure affecting the fatigue behavior of fuel cladding **tubes** was estd. to be 0.1 Pa. This value was much higher than the calcd. equil. vapor pressure of I in fuel rods, indicating that effects of I on the fatigue life would be very small or negligible during variable loading conditions. The n irradn. increased the fatigue life of cladding **tube** for the total strain amplitude >0.3% and decreased it <0.3%. The increase or decrease in fatigue cycles was attributed to the hardening effect or localized deformation in the irradiated material, resp. Fatigue limit of unirradiated Zry-2 **tubes** was 0.22%, and n irradn. reduced the value to 0.18%. The total strain amplitude of 0.18% coincided with the elastic strain at the proportional limit under the uniaxial tensile test of irradiated Zry-2. Both for unirradiated and irradiated specimens, transgranular fracture surfaces were induced by the bending. Ductile fracture surfaces were obsd. for unirradiated material, and n irradn. changed this surface into a typical brittle transgranular one.

CC 71-5 (Nuclear Technology)
Section cross-reference(s): 56

ST neutron irradn **Zircaloy** fuel cladding fatigue

IT **Pipes** and **Tubes**
(fatigue behavior of neutron-irradiated **Zircaloy**-2 fuel-cladding)

IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, fatigue behavior of neutron irradiated **Zircaloy**-2 **tube** BWR)

IT 12586-31-1, Neutron
RL: PROC (Process)
(fatigue behavior of **Zircaloy**-2 fuel cladding **tubes** bombarded by)

IT 7553-56-2, Iodine, properties
RL: PRP (Properties)
(fatigue behavior of **Zircaloy**-2 fuel cladding **tubes** neutron irradiated in presence of)

IT **11068-94-3, Zircaloy**-2
RL: PROC (Process)
(fatigue behavior of neutron-irradiated fuel cladding **tubes** of)

IT 12586-31-1
RL: PROC (Process)
(**nuclear** reactor **fuels** and **fuel** elements, **claddings**, fatigue behavior of neutron irradiated **Zircaloy**-2 **tube** BWR)

IT **11068-94-3, Zircaloy**-2
RL: PROC (Process)
(fatigue behavior of neutron-irradiated fuel cladding **tubes** of)

L67 ANSWER 23 OF 42 HCAPLUS COPYRIGHT 2002 ACS

1991:497493 Document No. 115:97493 **Manufacture** of zirconium alloy **tubes**. Harada, Makoto; Kanehara, Mitsuo; Abe, Katsuhiro (Kobe Steel, Ltd., Japan). Jpn. Kokai Tokkyo Koho JP 02270949 A2 19901106 Heisei, 4 pp. (Japanese). CODEN: JKXXAF. APPLICATION: JP 1989-93919 19890412.

AB Molten **Zr alloy** is extruded, hot-rolled and cold-rolled to give a high-strength **tube** esp. useful for **nuclear fuel clad**. The **tube** shows corrsion crack **resistance** and high **creep** rupture **strength**.
IC ICM C22F001-18
ICS B21B023-00
CC 56-11 (Nonferrous Metals and Alloys)
Section cross-reference(s): 71
ST zirconium alloy **tube** extrusion rolling
IT **Pipes** and **Tubes**
(zirconium alloy, extrusion and rolling of)
IT Zirconium alloy, base
RL: USES (Uses)
(**tubes**, extrusion and rolling of, for **nuclear fuel clad tubes**)

L67 ANSWER 26 OF 42 HCAPLUS COPYRIGHT 2002 ACS
1988:481675 Document No. 109:81675 Mechanical properties of zirconium alloy cladding **tubes** and critical fuel element power ramps. Novak, J.; Lauerova, D. (Inf. Cent., Nucl. Res. Inst., Rez, Czech.). Ustav Jad. Vyzk., [Rep.], UJV 8324 M, 12 pp. (English) 1988. CODEN: UJVYAK. ISSN: 0577-3857.
AB Study of the mechanism of cladding dehermetization under power ramp conditions **made** it possible to substantiate and to refine an empirical correlation between hoop fracture strain measured after internal pressurization .epsilon.f and the crit. power ramp value .DELTA.NC. Relations of conventional mech. properties, of fracture **toughness** in I atm. KISCC and of .DELTA.NC were found. In a present version, a correlation of .epsilon.f and .DELTA.NC is based on a semiempirical model having certain phys. interpretation. To quantify this correlation, it is now sufficient to know a single point, based on power ramp expts. in a MTR. Typical .DELTA.NC values for BWR and PWR confirm the model predictions and thus may be considered as verification of the proposed model. A more precise assessment of .DELTA.NC = 13 kW/m was obtained for WWER fuel elements with Zr1Nb cladding, irradiated to satn. at 300.degree.. A task to find .DELTA.NC for higher cladding temps. typical for WWER-440 and WWER-1000 reactors is thus transformed to the detn. of the cladding mech. properties after irradn. at corresponding temps.
CC 71-5 (Nuclear Technology)
Section cross-reference(s): 56
IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, mech. properties of zirconium alloy, verification of model for)
IT Zirconium alloy, base
RL: PROC (Process)
(mech. properties of cladding **tubes** of, verification of model for)
IT 12742-60-8, **Zircaloy**
RL: PROC (Process)
(mech. properties of cladding **tubes** of, verification of model for)

L67 ANSWER 28 OF 42 HCAPLUS COPYRIGHT 2002 ACS
1985:531028 Document No. 103:131028 Nuclear fuel element. Dodelier, Jacques; Melin, Philippe (Framatome et Cie., Fr.). S. African ZA 8406896 A 19850424, 10 pp. (English). CODEN: SFXXAB. APPLICATION: ZA 1984-6896 19840904. PRIORITY: FR 1983-14327 19830908.
AB A nuclear fuel element was designed, which has greater reliability and simplification of **prodn**. The element comprises an open support

tube for the stack over at least a portion of its length, sepg. the cladding from the stack and sepd. from the cladding by a radial interval of initial thickness sufficient to retard the coming into contact of the **tube** and the cladding upon the swelling of the pellets under irradn. The cladding in particular is **made** up of a **tube** of **Zr alloy** which has undergone a recrystn. treatment at 550-650.degree. so as to increase its **creep resistance** and **resistance** to the action of the coolant.

IC ICM G21C

CC 71-5 (Nuclear Technology)

ST reactor fuel cladding **tube**; zirconium alloy cladding **tube**

IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, zirconium alloy open-support **tube** for)

IT Zirconium alloy, base
RL: PROC (Process)
(**nuclear** reactor **fuel cladding** **tube** from, with open support)

L67 ANSWER 29 OF 42 HCAPLUS COPYRIGHT 2002 ACS

1985:208047 Document No. 102:208047 **Zircaloy tube** for **nuclear fuel cladding**. (Hitachi, Ltd., Japan). Jpn. Kokai Tokkyo Koho JP 60026650 A2 19850209 Showa, 4 pp. (Japanese). CODEN: JKXXAF. APPLICATION: JP 1983-132712 19830722.

AB To **make** corrosion- and **stress** corrosion cracking-**resistant**, an extruded **Zircaloy** 2 [**11068-94-3**] **tube** is quenched from .gtoreq.870.degree., cold rolled slightly to give precise inner diam., annealed at 600.degree. for 2 h, pickled in aq. HNO3-HF, heated at 500.degree., inserted with a Zr **tube**, drawn in vacuum, electron beam-welded at both ends, oil press expanded to weld, cold rolled, annealed at 590.degree., cut, machined to remove the Zr liner, and acid pickled. The wt. gain is .apprx.100 mg/dm2 and nodular corrosion neg. in a conventional corrosion test, compared to .apprx.1000 mg/dm2 and pos. for the **tube** **prepd**. by by a conventional process.

IC ICM C22F001-18
ICS B21C037-06; G21C003-20

CC 56-5 (Nonferrous Metals and Alloys)
Section cross-reference(s): 71

ST **Zircaloy tube nuclear fuel cladding**; thermomech treatment **Zircaloy tube**

IT **Pipes** and **Tubes**
(**Zircaloy**, thermomech. treatment of, for **nuclear fuel cladding**)

IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, thermomech. treatment of **Zircaloy** **tubes** for)

IT **11068-94-3**
RL: USES (Uses)
(thermomech. treatment of **tubes** of, for **nuclear fuel cladding**)

IT **11068-94-3**
RL: USES (Uses)
(thermomech. treatment of **tubes** of, for **nuclear fuel cladding**)

L67 ANSWER 30 OF 42 HCAPLUS COPYRIGHT 2002 ACS

1985:35009 Document No. 102:35009 Cladding **tube** for reactor fuel element. (Genshi Nenryo Kogyo K. K., Japan). Jpn. Kokai Tokkyo Koho JP 59131196 A2 19840727 Showa, 5 pp. (Japanese). CODEN: JKXXAF.

APPLICATION: JP 1983-5186 19830118.
AB In a **cladding tube** for **nuclear fuel** elements based on an **alloy** (e.g. **Zr alloy**) having a hcp. structure, the **tube** is obtained in such a manner that the central crystal axis of the hcp. alloy structures **makes** an angle of 0.degree., .+-.30.degree., and .+-.30.degree. with respect to the radius of the **tube** at the inner surface of the **tube** , at the interior of the **tube**, and at the exterior surface of the **tube**, resp. The **tube** is more **resistant** to **stress** corrosion cracking, hydriding and peripheral direction stretching.
IC G21C003-06; B21B019-10
ICA C22C016-00
CC 71-5 (Nuclear Technology)
Section cross-reference(s): 56
ST fuel element cladding **tube** alloy
IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, zirconium alloy **tubes**)
IT Containers
(**tubes**, zirconium alloy, for nuclear reactor fuel elements)
IT Zirconium alloy, base
RL: PROC (Process)
(**cladding tubes** of, for **nuclear fuel** elements)

L67 ANSWER 32 OF 42 HCAPLUS COPYRIGHT 2002 ACS
1983:97690 Document No. 98:97690 Fuel rods for nuclear reactors. (Toshiba Corp., Japan). Jpn. Tokkyo Koho JP 57042199 B4 19820907 Showa, 4 pp. (Japanese). CODEN: JAXXAD. APPLICATION: JP 1978-162490 19781228.
AB The cladding **tube** for **prepg**. reactor fuel rod is obtained by depositing a Cu layer on the inner wall of a **Zr alloy tube**, oxidizing the Cu layer, then reducing the oxidized Cu layer. The pellet-cladding interaction is reduced and I-absorption capability is increased. The rod is useful in a LWR and is **resistant** towards **stress** corrosion cracking.
IC G21C003-06
CC 71-5 (Nuclear Technology)
ST copper coating **Zircaloy** cladding fuel; LWR fuel cladding copper layer; reactor fuel cladding copper layer
IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, copper coated zirconium alloy, for reduced stress corrosion cracking and pellet-cladding interaction)
IT Zirconium alloy, base
RL: PROC (Process)
(**nuclear** reactor **fuel** element **cladding tube** of, copper coating for reduced stress corrosion cracking and fuel pellet-cladding interaction)
IT 7440-50-8, uses and miscellaneous
RL: USES (Uses)
(**nuclear** reactor **fuel** elements **cladding tube** of zirconium alloy coated by, for reduced stress corrosion cracking and pellet-cladding interaction)

L67 ANSWER 33 OF 42 HCAPLUS COPYRIGHT 2002 ACS
1983:61692 Document No. 98:61692 Iodine-induced stress corrosion cracking of copper-barrier **Zircaloy-4 tubes**. Huang, Jenn Hwa; Lee, Tien; Chuang, Yii Der (Inst. Nucl. Energy Res., Lung-Tan, Taiwan). Ts'ai Liao K'o Hsueh, 13(1), 6-12 (English) 1981. CODEN: TLKHAJ. ISSN: 0379-6906.
AB The effect is discussed of vacuum-evapd. and electroplated thin Cu layers

on the stress corrosion susceptibility of barrier-type **Zircaloy-4** **[11068-95-4]** **tubes**. Pressurization tests were run on specimens from a single batch of **Zircaloy** **tubing** with various thickness of Cu layers at .apprx.573 and .apprx.633 K, resp. Specimens were internally pressurized with Ar contg. a nominal I concn. of 5 mg/cm2 of **Zircaloy** surface. The times-to-failure of the Cu-coated specimens were markedly longer as compared to those of the uncoated ref. specimens. No crack was obsd. on Cu films at **stresses** below the burst **strength** of the **tubes**. The Cu film reacted with I after extensive exposure in I vapor, and a brittle **product** was **formed** which might reduce the protectiveness of this plated layer. Though cleaner than the electroplated Cu film, the vacuum evapd. film was less compatible with **Zircaloy** **tubes** when its thickness exceeded a few microns. The higher purity of the vacuum-evapd. film did not benefit very much its stress corrosion cracking protectiveness.

CC 71-5 (Nuclear Technology)
Section cross-reference(s): 56

ST iodine stress corrosion cracking **Zircaloy**; fuel cladding **Zircaloy** iodine cracking; copper barrier **Zircaloy** iodine cracking; reactor fuel cladding copper barrier

IT **Pipes** and **Tubes**
(iodine-induced stress corrosion cracking of copper-barrier **Zircaloy-4**)

IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, iodine-induced stress corrosion cracking of copper-barrier **Zircaloy-4**)

IT 7440-50-8, reactions
RL: RCT (Reactant)
(iodine-induced stress corrosion cracking of **Zircaloy** **tubes** with barrier layer of)

IT **11068-95-4**
RL: PROC (Process)
(iodine-induced stress corrosion cracking of copper-barrier **tubes** of)

IT 7553-56-2, reactions
RL: RCT (Reactant)
(stress corrosion cracking of copper-barrier **Zircaloy** **tubes** induced by)

IT **11068-95-4**
RL: PROC (Process)
(iodine-induced stress corrosion cracking of copper-barrier **tubes** of)

L67 ANSWER 34 OF 42 HCAPLUS COPYRIGHT 2002 ACS

1982:604566 Document No. 97:204566 Beta-quenching of **Zircaloy** cladding **tubes** in intermediate or final size - methods to improve corrosion and mechanical properties. Andersson, T.; Vesterlund, G. (Sandvik AB, Sandviken, Swed.). ASTM Spec. Tech. Publ., 754(Zirconium Nucl. Ind.), 75-95 (English) 1982. CODEN: ASTTA8. ISSN: 0066-0558.

AB Three batches of **Zircaloy**-2 **[11068-94-3]** **tubing** were .beta.-quenched prior to the final cold-rolling, cold rolled 80 %, and annealed at 475 - 575.degree.. A 4th batch was .beta.-quenched in the final size. For comparison, std. **tubing** was included in all tests performed. The 2nd-phase particles were studied by means of optical and SEM. Corrosion testing was carried out at 400.degree. and in high-temp. (475 - 500.degree.) high-pressure steam. The mech. tests comprised tension, burst, and creep testing under internal pressure. .beta.-Quenching instead of an intermediate or the final anneal results in significant structural changes. The most striking features are

the **formation** of a structure consisting of plates of .alpha.-phase and the repptn. of much finer 2nd-phase particles in the plate boundaries. Cold-rolling of .beta.-quenched hollows followed by a final anneal in the .alpha.-range will give an equiaxed structure, but the size and distribution of the 2nd phase obtained in .beta.-quenching will not be markedly changed. The wt. gain at 400.degree. increases slightly as a result of .beta.-quenching in intermediate or final size. In high-pressure steam at 475 - 500.degree., on the other hand, such .beta.-quenching has a dramatic beneficial effect on the corrosion resistance. The short-term strength as measured in tension and burst testing is improved by .beta.-quenching of hollows or finished **tubes**, whereas such treatment results in a slight drop in ductility, esp. for **tubing** .beta.-quenched in the final size. The 400.degree. transverse **creep strength** is increased by the introduction of .beta.-quenching prior to the final cold-rolling. The improvement is caused mainly by small 2nd-phase particles, **formed** during .beta.-quenching, which gives rise to pptn. hardening.

CC 71-5 (Nuclear Technology)

ST **Zircaloy** cladding beta quenching; reactor fuel cladding corrosion prevention

IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, **Zircaloy**, .beta.-quenching of, for improved corrosion prevention and mech. properties)

IT **11068-94-3**
RL: PROC (Process)
(.beta.-quenching of cladding **tubes** of, for improved corrosion prevention and mech. properties)

IT **11068-94-3**
RL: PROC (Process)
(.beta.-quenching of cladding **tubes** of, for improved corrosion prevention and mech. properties)

L67 ANSWER 38 OF 42 HCAPLUS COPYRIGHT 2002 ACS

1981:111072 Document No. 94:111072 Behavior of unirradiated zirconium-lined and copper-plated **Zircaloy-2 tubing** under simulated PCI conditions. Gangloff, R. P. (Corp. Res. Dev. Dep., Gen. Electr. Co., Schenectady, NY, USA). Report, GEAP-25093, 109 pp. Avail. INIS; NTIS From: INIS Atomindex 1980, 11(23), Abstr. No. 565953 (English) 1979.

AB The expanding mandrel technique was used to evaluate the **fracture resistance** of unirradiated, cold-worked Zr and Cu barrier fuel cladding for simulated pellet-cladding interaction (PCI) conditions. A range of environment and loading conditions, shown to **produce** severe embrittlement of cold-worked **Zircaloy**-2 [**11068-94-3**] and including 4 Pa I2, pure Cd and molten Cd-satd. Cs, was employed for barrier screening.

CC 71-6 (Nuclear Technology)
Section cross-reference(s): 56

ST reactor fuel cladding embrittlement **Zircaloy**; BWR fuel cladding embrittlement **Zircaloy**; **fracture resistance** zirconium

IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, embrittlement of **Zircaloy**, fuel-cladding interactions in relation to)

IT 7440-43-9, uses and miscellaneous
RL: USES (Uses)
(as barrier screening for **Zircaloy** reactor fuel claddings)

IT 7440-46-2, uses and miscellaneous
RL: USES (Uses)
(barrier screening from cadmium-satd., for **Zircaloy** fuel

claddings, embrittlement in relation to)
IT 7553-56-2, uses and miscellaneous
RL: USES (Uses)
(embrittlement of **Zircaloy** exposed to fission-**product**
)
IT **11068-94-3**
RL: PROC (Process)
(embrittlement of reactor fuel claddings of)
IT 7440-67-7, properties
RL: PRP (Properties)
(**fracture resistance** of unirradiated cold-worked)
IT **11068-94-3**
RL: PROC (Process)
(embrittlement of reactor fuel claddings of)

L67 ANSWER 39 OF 42 HCAPLUS COPYRIGHT 2002 ACS
1977:162543 Document No. 86:162543 Design of an irradiation device for the determination of the in-pile creep behavior of **Zircaloy** cladding **tubes** under internal and external overpressure, in FRG-2. Ahlf, J.; Reymann, A.; Eichhorn, O.; Gaertner, M. (Ges. Kernenergieverwert. Schiffbau Schiffahrt m.b.H., Geesthact, Ger.). J. Nucl. Mater., 65(1), 302-6 (English) 1977. CODEN: JNUMAM.
AB The dimensional stability of fuel rods in light-water reactors is influenced by the **creep strength** of the **Zircaloy** cladding. Irradn. expts. were **made** in FRG-2 to det. the effect of n irradn. on the creep behavior of **Zircaloy** cladding. In these capsule expts. specimens can be tested in a He environment at temps. of 280-400.degree. in a fast n flux of .apprx.5 .times. 1013/cm2-s under biaxial tensile and compressive stresses of 70-150 N/mm2. The test equipment, the exptl. techniques, and the initial results are described.
CC 71-5 (Nuclear Technology)
Section cross-reference(s): 56
ST app creep **Zircaloy** cladding
IT **Nuclear** reactor **fuels** and **fuel** elements
(**claddings**, creep behavior of **Zircaloy**, irradn. device for detn. of in-pile)
IT **11068-95-4** 12742-60-8
RL: PROC (Process)
(cladding **tubes**, in-pile creep behavior of, irradn. device for detn. of)
IT 12586-31-1, chemical and physical effects
RL: PEP (Physical, engineering or chemical process); PROC (Process)
(on creep behavior of **Zircaloy** cladding **tubes**, irradn. device for detn. of in-pile)
IT **11068-95-4**
RL: PROC (Process)
(cladding **tubes**, in-pile creep behavior of, irradn. device for detn. of)

L67 ANSWER 40 OF 42 HCAPLUS COPYRIGHT 2002 ACS
1976:156987 Document No. 84:156987 Creep anisotropy of **Zircaloy** cladding **tubes**. Stehle, H.; Steinberg, E. (Kraftwerk Union A.-G., Erlangen, Ger.). Report, AED-CONF-75-170-004, 12 pp. Avail. INIS From: Nucl. Sci. Abstr. 1976, 33(4), Abstr. No. 7808 (German) 1975.
AB Survey is given on the texture of **Zircaloy** [12742-60-8] cladding **tubes** obtained depending on the **manufacturing** conditions, and the state of knowledge on the anisotropy of the mech. properties of the zirconium alloys connected with the texture is outlined. Theor. formulations are set up for the phenomenol. representation of the

anisotropic creep. The results of tension and compression tests and the thus obtained creep site curves exhibit distinct differences with **tubes** having different textures. Furthermore, an asymmetry regarding compressive tensile stress is found in such a manner that the material under compression **stress** is more **resistant** to **creep**. Discussions follow on the deformation mechanisms, and a comparison with flow processes as well as indications on the significance of these creep results within the framework of fuel rod design are given.

CC 71-5 (Nuclear Technology)
Section cross-reference(s): 56

ST **Zircaloy** cladding **tube** creep; fuel cladding **Zircaloy**

IT **Nuclear** reactor **fuels** and **fuel** elements
(**cladding tubes** for, creep anisotropy of **Zircaloy**)

IT **Pipes** and **Tubes**
(creep anisotropy of **Zircaloy** cladding)

IT 12742-60-8
RL: PROC (Process)
(creep anisotropy of **cladding tubes** of, for **nuclear** reactor **fuels**)

L67 ANSWER 41 OF 42 HCAPLUS COPYRIGHT 2002 ACS

1971:483059 Document No. 75:83059 **Fabrication** technology and quality for **Zircaloy** fuel-cladding **tubes**. Yamamoto, Haruo; Okada, Takeshi; Nagai, Nobuyuki; Tanaka, Yoshiro (Kobe Steel, Ltd., Kobe, Japan). Karyoku Hatsuden, 21, 217-22 From: Nucl. Sci. Abstr. 1971, 25(10), 22041 (Japanese) 1970.

AB The quality of domestic **Zircaloy** fuel-cladding **tubes** was improved. The features of the established domestic technol. and quality control are described. The requirements for cladding **tubes**, in general, and the **production** procedure for **Zircaloy tubes** are explained. The **fabrication** of the **Zircaloy** cladding **tubes** are accomplished by cold-working. To improve the quality, it was necessary to have corrosion-**produced** Zr hydride in the circumferential direction. The 3 techniques, draw bench, pilger roll, and 3-roller roll, and the corresponding hydride behaviors, are described. The 3-roller rolling was best for the hydride problem and resulted in superior surface, dimensional accuracy, strength, and **toughness** properties.

CC 76 (Nuclear Technology)

ST **fabrication Zircaloy** fuel cladding **tubes**;
rolling **Zircaloy** fuel cladding **tubes**

IT **Nuclear** reactor **fuels**, uses and miscellaneous
(**claddings**, **Zircaloy**)

IT **Zircaloy**
(**nuclear** reactor **fuel claddings**)

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1971:483057 Document No. 75:83057 **Fabrication** techniques and quality of **Zircaloy** cladding **tubes**. Yamamoto, Haruo; Okada, Takeshi; Onishi, Tadatoshi; Tanaka, Yoshiro (Kobe Steel, Ltd., Kobe, Japan). Kobe Seiko Giho, 20(2), 12-20 From: Nucl. Sci. Abstr. 1971, 25(10), 22039 (Japanese) 1970.

AB The **Zircaloy** cladding **tubes produced** by Kobe Steel Ltd. are superior in quality compared with those of other countries. The requirements for **Zircaloy** cladding **tubes** and the procedure adopted by Kobe Steel Ltd. are described, and various properties of the **produced tubes** are given. The requirements included corrosion resistance, compatibility with contained fuel, and

strength against the internal pressure of fission **product** gas. The overall procedure consisted of hot extrusion to give high purity and uniformity, cold processing to provide accurate dimensions, good **toughness**, and finally inspection. A 3-roller method used in the final process gave the highest difference between the redn. ratios of wall-thickness and outerdiam.

CC 76 (Nuclear Technology)

ST **fabricating Zircaloy** cladding fuel **tubes**; rolling **Zircaloy** cladding fuel **tubes**; extrusion **Zircaloy** cladding fuel **tubes**

IT **Nuclear** reactor **fuels**, uses and miscellaneous
(**claddings, Zircaloy**)

IT **Zircaloy**
(**nuclear** reactor **fuel claddings**)